Safety Guide 100

DESIGN GUIDE FOR PACKAGING AND OFFSITE TRANSPORTATION OF NUCLEAR COMPONENTS, SPECIAL ASSEMBLIES, AND RADIOACTIVE MATERIALS ASSOCIATED WITH THE NUCLEAR EXPLOSIVES AND WEAPONS SAFETY PROGRAM

CHAPTER 5.0

RADIATION SHIELDING ASPECTS

November 7, 1994

MARTIN MARIETTA ENERGY SYSTEMS, INC.

Oak Ridge, TN 37830

THIS PAGE INTENTIONALLY LEFT BLANK

CONTENTS

FIGURES

TABLES

THIS PAGE INTENTIONALLY LEFT BLANK

ACRONYMS

ALARA	As Low As Reasonably Achievable
CFR	Code of Federal Regulations
CG	Combinatorial Geometry
DOE	Department of Energy
DOT	Department of Transportation
GP	Geometric Progression
HAC	Hypothetical Accident Condition
IAEA	International Atomic Energy Agency
ICRU	International Commission of Radiation Units and Measurements
LANL	Low Alamos National Laboratory
NCT	Normal Conditions of Transport
NRC	Nuclear Regulatory Commission
RSIC	Radiation Shielding Information Center
SSR	Safe-Secure Railcar
SST	Safe-Secure Trailor
TI	Transport Index

THIS PAGE INTENTIONALLY LEFT BLANK

DEFINITIONS

Al - The maximum activity of special form radioactive material permitted in a Type A package.

A2 - The maximum activity of radioactive material, other than special form radioactive material, permitted in a Type A package.

Carrier - A person engaged in the transportation of passengers or property by land or water as a common, contract, private carrier, or by civil aircraft.

Components - Nuclear parts and hazardous parts that comprise and/or are associated with the nuclear weapons program. Nuclear components - Nuclear weapon parts that contain fissile and/or radioactive materials. Hazardous components - Nuclear weapon parts that contain hazardous materials other than fissile and/or radioactive materials.

Containment System - The components of the packaging intended to retain the radioactive material during transport.

Contractor - A contractor managing or operating government-owned or -leased property on behalf of the Department of Energy.

Conveyance - Any vehicle, aircraft, vessel, freight, container, hold compartment, or defined deck area of an inland waterway craft or seagoing vessel.

DOE and NRC Certificate of Compliance - A certificate issued by DOE or NRC, as appropriate, approving for use, with specified limitations, a specific packaging for quantities of radioactive materials exceeding A1/A2 quantities as defined in DOE and NRC regulations.

DOE transport - Conveyance within a DOE-owned transportation system (e.g., Safe-Secure Trailer (SST), Safe-Secure Railcar (SSR), and/or government-owned aircraft and vehicles).

Dose Equivalent - A quantity used for radiation protection that expresses on a common scale for all radiations the irradiation incurred by exposed persons; the product of the absorbed dose, the quality factor, and any other modifying factors. (The rem and the sievert are the units of dose equivalent.)

Dose Rate - The radiation dose delivered per unit time; measured in rem per hour or other equivalent units.

Equivalent protection - Alternative measures that will achieve a level of safety at least equal to that specified in the regulations from which the alternative is sought, which will be consistent with the public intent and will provide adequate protection against risks to life and property.

Exclusive use (also referred to in other regulations as “sole use” or “full load”) - The sole use of a conveyance by a single consignor and for which all initial, intermediate, and final loading and unloading are carried out in accordance with the direction of the consignor or consignee. Exclusive use applies to transport by SST and SSR.

Fissile classification - The categorization of fissile material packages into one of the following three classes according to the controls needed to provide nuclear criticality safety during transportation.

1. **Fissile Class I** - A package that may be transported in unlimited numbers and in any arrangement and that requires no nuclear criticality safety controls during transportation. A transport index is not assigned for nuclear criticality safety but may be required because of external radiation levels.

2. **Fissile Class II** - A package that may be transported together with other packages in any arrangement but, for criticality control, in numbers that do not exceed an aggregate transport index of 50. These shipments require no other nuclear criticality safety control during transportation. Individual packages may have a transport index of not less than 0.1 and no more than 10.

3. **Fissile Class III** - A shipment of packages that is controlled by specific agreement between the shipper and the carrier to provide nuclear criticality safety.

Note: The proposed revision of 10 CFR 71 eliminates the use of the fissile classes.

Fissile material - Any material consisting of or containing one or more fissile radionuclides.

Fissile radionuclides - Uranium-233 and ^{235}U, and ^{238}PU, ^{239}Pu, and ^{241}Pu, or any combination of these radionuclides, including trace amounts of higher actinides. Unirradiated natural uranium or depleted uranium and natural or depleted uranium that has been irradiated only in thermal reactors are not included in this definition.

Hazardous material - A substance or material that the Secretary of Transportation has determined to be capable of posing an unreasonable risk to health, safety, and property when transported in commerce and that has been so designed.

Maximum normal operating pressure - The maximum gauge pressure that would develop in the containment system in a period of one year under the heat test specified in 10 CFR 71.71(c)(1) in the absence of venting, external cooling by an ancillary systems, or operational controls during transport.

Neutron Poisons - Materials other than fissile material that will absorb neutrons, especially materials such as boron.

Normal form radioactive material - Radioactive material that has not been demonstrated to qualify as special form radioactive material.

Optimum interspersed hydrogenous moderation - The presence of hydrogenous material between packages to such an extent that the maximum nuclear reactivity results.

Package - The packaging and its radioactive contents as presented for transport.

1. **Fissile material package** - A fissile material packaging together with its fissile contents.

2. **Type B package** - A Type B packaging and its radioactive contents. On approval, Type B package design is designated by NRC or DOE as B(U) unless the package has a maximum normal operating pressure of more than 700 kpa (100 lb/in^2) gauge or a pressure relief device that would allow the release of radioactive material to the environment under the tests specified in 10 CFR 71.73 (hypothetical accident conditions), in which case it will receive a designation B(M). B(U) refers to the need for unilateral approval of international shipments; B(M) refers to the need for multilateral approval. No distinction made in how packages with these designations may be used in domestic transportation. To determine their distinction for international transportation, see DOT regulations in 49 CFR 173. A Type B package approved before September 6, 1983, was designated only as Type B. Limitations on its use are specified in 10 CFR 71.13.

Packaging - The assembly of components necessary to ensure compliance with the packaging requirements of 10 CFR 71 or DOE Orders 5610.12 (Draft). It may consist of one or more receptacles, absorbent materials, spacing structures, thermal insolation, radiation shielding, and devices for cooling or absorbing mechanical shocks. The vehicle, tie-down system, and auxiliary equipment may be designated as part of the packaging.

Quality assurance - Planned and systematic action necessary to provide adequate confidence that a facility, structure, system, or component will perform satisfactorily and safely in service. The goal of quality assurance is to ensure that research, development, demonstration, scientific investigations, and production activities are performed in a controlled manner; that components, systems, and processes are designed, developed, constructed, tested, operated, and maintained according to engineering standards, quality practices, and Technical Specifications/Operational Safety Requirements; and that resulting technology data are valid and retrievable. Quality assurance includes quality control, which comprises all actions necessary to control and verify the features and characteristics of a material, process, product, or service to specified requirements.

Quality assurance plan - A document that contains or references the quality assurance elements established for an activity, group of activities, scientific investigation, or project. It describes how conformance with such requirements is to be ensured for structures, systems, computer software, components, and their operation commensurate with 1) the scope, complexity, duration, and importance to satisfactory performance; 2) the potential impact on environment, safety, and health; and 3) requirements for reliability and continuity of operation.

Quality assurance program - A systematic program of controls and inspections applied by any organization or body involved in the transport of radioactive material to provide adequate confidence that the standard of safety prescribed in regulations is achieved in practice.

Quality factor - A multiplying factor used with absorbed dose to express dose equivalent. Its value is 1 for gamma rays and varying between 1 and 11 for neutrons according to the neutron energy.

Rad - A unit of absorbed dose. The word comes from the acronym radiation absorbed dose, and it is equivalent to 100 ergs/gram. It does not take into account the biological effect resulting from the absorbed dose.

Radioactive material - Any material having a specific activity greater than 0.002 microcuries per gram (mCi/g) that is to be used for the fabrication of nuclear components for nuclear weapons and/or nuclear test devices.

Rem - A unit of dose equivalent. The word comes from the acronym, roentgen equivalent man and takes into account the biological effect from an absorbed dose of radiation.

Roentgen - The unit for exposure. It is that amount of gamma or X rays required to produce ions carrying one electrostatic unit of electrical charge in 1 cm^3 of dry air under standard conditions.

Safety Analysis Report (SAR) - Formal documentation that systematically describes a system and that identifies and assesses associated hazards and/or risks for the purpose of demonstrating adequate safety.

Safety Analysis Report for Packaging (SARP) - A document that provides a comprehensive evaluation of the container and its contents to demonstrate safety compliance in accordance with DOE Order 5610.12 (Draft).

Safety Evaluation Report (SER) - A document that provides the evaluation of and recommendations by the review team of the SAR supporting the request for certification.

Safety Evaluation Report for Packaging (SERP) - A document that provides evaluation of and recommendations by the review team of SARP for the package design.

Special assemblies - Major assemblies of nuclear weapon components that do not comprise a complete nuclear explosive and, therefore, are incapable of producing a nuclear detonation.

Safe-secure railcar (SSR) - A specially designed railcar that has protective and deterrent systems that are used in a special train to transport nuclear explosives or special nuclear materials.

Safe-secure trailer (SST) - A specially designed semitrailer that has protective and deterrent systems that are used with a special tractor to transport nuclear explosives or special nuclear materials.

Sievert (Sv) - International unit dose equivalent, 1 Sv = 100 rem.

Special Form Radioactive Material - Radioactive material that satisfies the following conditions:

1. It is either a single solid piece or is contained in a sealed capsule that can be opened only by destroying the capsule.

2. The piece or capsule has at least one dimension not less than 5 mm (0.197 inch).

3. It satisfies the test requirements of 10 CFR 71.75.

5.0 RADIATION SHIELDING ASPECTS

5.1 INTRODUCTION

The radiation shielding aspects of the weapon components and special assembly packaging design guides are concerned with establishing that the radiation dose rate limits on the package exterior are not exceeded. The Department of Energy (DOE) requires the application of relevant federal regulations to ensure the protection of the public safety and health and the environment from the inherent risks of the public transportation of nuclear weapon components, special assemblies, and radioactive material.

The purpose of this design guide is to aid in the identification and efficient resolution of any radiation shielding issues arising from the public domain transportation of radioactive material associated with that portion of the U. S. nuclear weapons program under the control of DOE. This guide supports the shipment of Type B quantities of dispersible forms of radioactive material and special nuclear material.

5.2 RADIOACTIVE MATERIALS AND SOURCES

The radioactive materials addressed in this guide are plutonium, uranium, thorium, and tritium. Only unirradiated source material (i.e., material that has not been exposed to an operating nuclear reactor) is considered. An exception to this requirement is whenever any such irradiated material has concentrations measured in parts per million (ppm) or less by weight as compared with the unirradiated material or when the extent of exposure and/or decay time since exposure is such that the induced radioactivity is comparable to or lower than the natural radioactivity of the other package materials.

Only neutron and gamma radiation are considered in the shielding aspects of the package design and analysis. Except as noted in the preceding paragraph, this radiation is due primarily to the radioactive decay and spontaneous fission of the material. All α and β radiation from these processes are assumed to be absorbed in the package materials. (This absorption may contribute significantly to the internal heat generation for the package.) There may be a significant neutron source, relative to spontaneous fission, from (α, n) reactions with nonradioactive compound materials and trace element impurities in the radioactive material. To a lesser extent, there may be an (α, n) neutron source from a surface effect when an α decay radioactive material has a large, common surface with a low-Z material with a large (α, n) cross section. A large source of low-energy gamma rays may be present due to the Bremsstrahlung radiation from β decay, but most low-energy gammas are attenuated by the package material or self-shielding of the package contents. Induced fission neutrons and gamma rays in fissile material, resulting from spontaneous fission neutrons, should be included in either the source definition or any subsequent analysis. The dose rate exterior to the package resulting from induced fissions should be small due to the degree of subcriticality of the fissile material. Any fissile material in the package will be under strict criticality control independent of the shielding considerations. There will also be a small source of secondary gamma rays from neutron interactions with the source and other package materials.

Other possible sources (such as photoneutrons, activation neutrons and gamma rays, etc.) can be neglected. Any significant radiation sources that contribute to the package external dose rates will be accounted for in the source and transport computer codes that are described later. The source materials of interest are as follows.

5.2.1 Plutonium

Weapons-grade plutonium will consist of more than 90% ^{239}Pu by weight with a few percent ^{240}Pu; fractions of a percent of ^{238}Pu, ^{241}Pu, and ^{242}Pu will also be present. The ^{236}Pu isotope at concentrations on the order of parts per billion (ppb) can lead to a significant gamma ray source due to the 2.6 MeV gamma ray from ^{208}Tl. The decay of ^{241}Pu can lead to a large gamma ray source from ^{241}Am and, to a lesser extent, from ^{237}U. The presence of fluorine, boron, lithium, and beryllium even in trace concentrations in the plutonium can lead to a significant (α, n) source relative to spontaneous fission in plutonium. Oxygen, carbon, and other trace or compound elements can also contribute to an (α, n) source when included with plutonium.

All of the isotopes mentioned in the preceding paragraph can contribute to the shipping package exterior dose rate depending on the isotope concentrations, the packaging materials, and the decay time since production of the plutonium. The times of maximum dose rate can vary from a few years to several hundred years, depending on these factors. The longer times become significant when the shipping package is also to be used for the long-term storage of plutonium. On a per nuclide basis, the major gamma ray sources are from the decay chains of ^{236}Pu, ^{238}Pu, and ^{241}Pu, and the major neutron sources are from ^{238}Pu, ^{240}Pu, and ^{242}Pu. Only when the plutonium is almost entirely ^{239}Pu is this isotope a major contributor to the dose rate. For the analysis of a plutonium package with a range of possible isotopic concentrations and conservatively subcritical, a conservative shielding model would normally include the least possible amount of ^{239}Pu and the maximum amount of ^{236}Pu, ^{238}Pu, and ^{241}Pu. Transportation packages containing plutonium can easily have exterior dose rates that are a significant fraction of the regulatory limits, and overly conservative calculational models of these packages may compute dose rates that exceed the limits. A sample problem with plutonium is analyzed in Subsect. 5.8.2, and several of the items mentioned in this section are examined in more detail there.

5.2.2 Uranium

Highly enriched uranium is primarily ^{235}U with a few percent by weight of ^{238}U. Natural or depleted uranium is almost entirely ^{238}U with a fraction of a percent ^{235}U. The ^{234}U isotope will be present in concentrations of about one percent in weapons uranium. The 23% isotope is sometimes present in trace amounts, but the ^{233}U and ^{237}U isotopes will not be considered.

The gamma ray dose rate on the exterior of a package containing only uranium as a source material will generally be a small fraction of the regulatory limit. However, the ^{232}U isotope can sometimes be found in enriched uranium in trace amounts, and at concentrations of several parts per billion the gamma ray dose rate increases significantly due to the 2.6 MeV gamma ray from ^{208}Tl. But even reasonably conservative calculational shielding models of a shipping package containing uranium with trace amounts of U^{232} should give dose rates much less than the regulatory limits. The maximum dose on the exterior of a uranium package with ^{232}U will occur about 10 years after fabrication of the pure uranium.

The radioactive decay of unirradiated, conservatively subcritical uranium produces very little neutron dose rate. Very conservative calculational models of a uranium package will produce neutron and secondary gamma ray dose rates at or below background levels. Some trace elements or compound materials may increase the neutron source due to (α, n) over that from spontaneous fission, but the resulting dose rates are still very low (on the order of a few mrem/h at most).

5.2.3 Thorium

Natural thorium is almost entirely ^{232}Th. The decay products existing in trace concentrations can lead to a significant gamma ray dose from the 2.6 MeV gamma ray from ^{208}Tl, and thorium should be included in the analysis of the package of which it is a constituent. The neutron generation properties of thorium decay are less than that for uranium. The maximum activity from ^{232}Th decay occurs at about 50 years after fabrication of pure thorium.

5.2.4 Tritium

The radioactive decay of tritium produces no significant source of neutrons or gamma rays, and this isotope can be neglected in the shielding considerations for shipping packages.

5.3 PACKAGE MODEL

The regulations governing the permissible radiation levels exterior to a shipping package are best explained relative to a generic package design shielding model, which is shown in Fig. 5.1 as a series of concentric cylinders. The centrally located contents of variable shape or shapes include the radioactive source material. The radioactive source and any other contents are surrounded by semi-rigid, resilient foam packing material for uranium. Plutonium packing will, in general, not contain hydrocarbons or like material. Both the contents and the packing material may contain voids or other geometric irregularities. The inner container, usually of stainless steel, provides the primary containment boundary for the package contents. The inner container may contain irregularities, such as a top lid that is thicker than the side walls or bottom, or it may consist of upper and lower sections joined at an axially central location. The

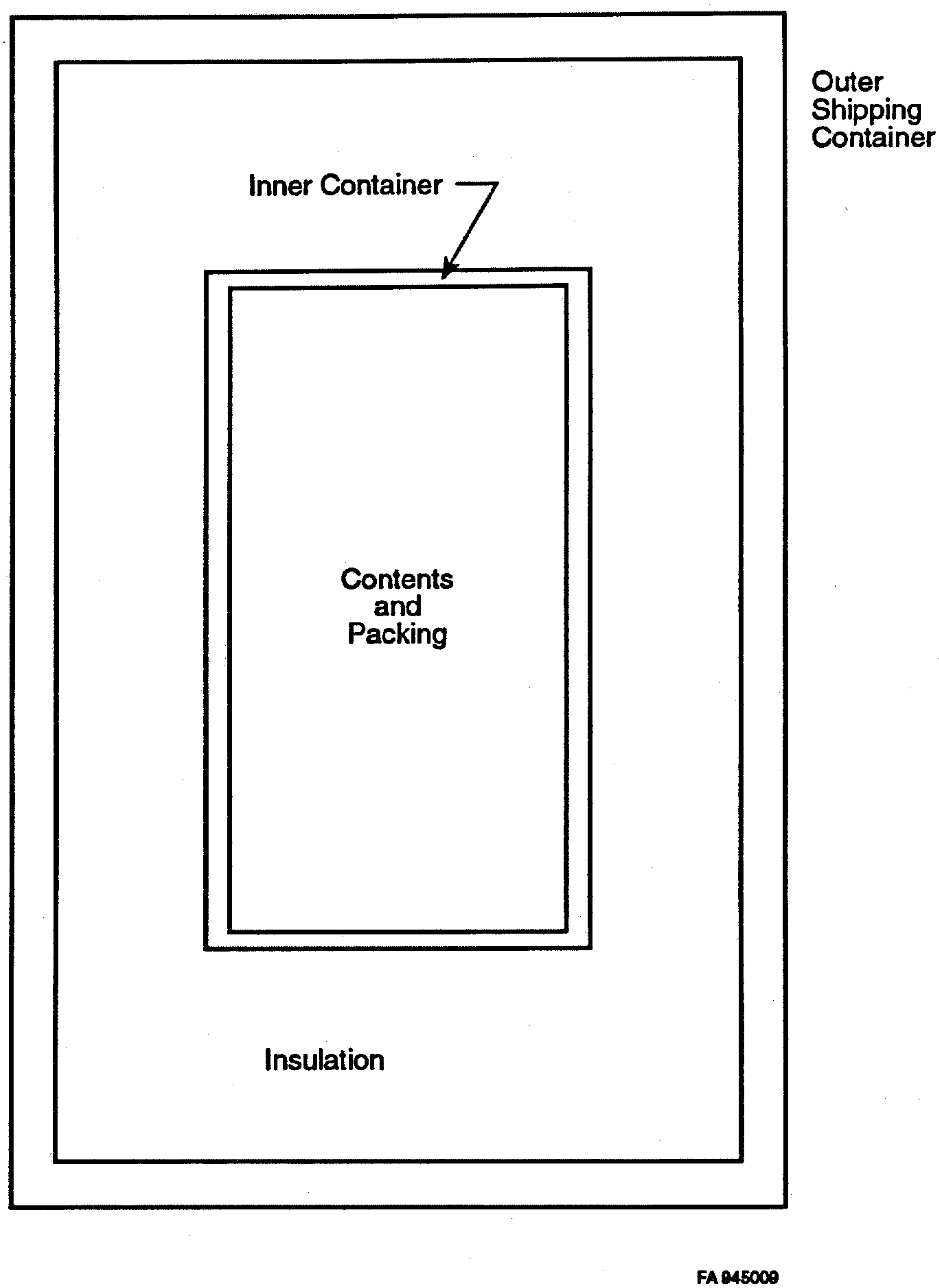

Fig. 5.1. Generic shipping package.

inner container may have a convex top or bottom, which could be modeled as flat at the greatest extent or at an average axial position of the curved surface.

The inner container is surrounded by rigid packing and/or insulation material with few voids or other irregularities. The outer container is a thin-walled shipping drum of carbon steel or stainless steel. Throughout the shipping package there may be support structures, flanges, fasteners, tie-downs, or lifting devices that can be conservatively omitted in the shielding model. The placement of the contents and the inner container will be such that the package center of gravity is below the axial mid-point of the outer shipping drum.

5.3.1 Normal Conditions of Transport

The preceding description of the model of the shipping package is for Normal Conditions of Transport (NCT).

The detector locations for maximum surface dose rates will ordinarily be on the axis at the top and bottom drum surface and at or near the vertical mid-point elevation of the source material on the drum side. The off-surface detector locations will be one meter from the corresponding surface detectors.

5.3.2 Hypothetical Accident Conditions

The Hypothetical Accident Conditions (HACs) are a set of proposed conditions relating to the package under various accident scenarios. In addition to shielding, other aspects of the package that are included are criticality, structural integrity, containment (for both the contents escaping from the package or outside material entering), and thermal conditions from an external heat source (internal heat generation is included under NCT). From the package requirements relating to these other aspects, one can conservatively conclude that the shielding model for HAC is the same as NCT with the insulation/packing material and outer shipping drum removed. The inner container, inner packing material, and contents remain intact, and the exterior package surface for HAC is the outer surface of the inner container. The HAC off-surface (one meter) detectors are defined relative to the inner container.

5.4 DOSE RATE LIMITS AND OTHER REGULATIONS

The Shielding Aspects of the Weapon Components and Special Assembly Packaging Design are governed by DOE Order 5610.12.[1] These documents specify the applicable federal regulations and restrictions that must be met, or exceeded, for the transportation of radioactive material. Specifically, the *Code of Federal Regulations* (CFR)[2] (10 CFR 71.49) gives the following dose rate limits.

5.4.1 Maximum Surface Dose Rate of 200 mrem/h for NCT

A package containing radioactive material must be designed and prepared for shipment for NCT such that the radiation level does not exceed 200 mrem/h at any point on the external surface of the package, as specified by 10 CFR 71.49, which is the applicable regulation for the Nuclear Regulatory Commission (NRC). The same limit is also specified for the U. S. Department of Transportation (DOT) in 49 CFR 173.410 and for the International Atomic Energy Agency (IAEA) in IAEA Safety Series No. 6.[3]

5.4.2 Maximum 1 Meter Dose Rate of 10 mrem/h for NCT

The maximum dose at one meter from any external surface position of the package for NCT must not exceed 10 mrem/h, as specified in 10 CFR 71.49, using the definition of Transport Index (TI) in 10 CFR 71.4.

5.4.3 Maximum 1 Meter Dose Rate of 1000 mr/hr for HAC

The maximum dose at one meter from any external surface position of the package for HAC must not exceed one rem per hour (1000 mrem/h), as specified in 10 CFR 71.51 (a2). For HAC, the dose of the external package surface is assumed to be that of the inner container (see Subsect. 5.3.2).

5.4.4 Exclusive-use Conveyance, Long-term Storage, and ALARA

The NCT dose rate limits apply to a shipping package without regard to the method of shipment. If the package is shipped as exclusive use, the NRC limits can be relaxed to take into account the material and geometric shielding properties of the conveyance vehicle (see 10 CFR 71.4 for the definition of "exclusive use"). A maximum package external dose of 1000 mrem/h for NCT is allowed in a closed vehicle if the 200 mrem/h limit is met on the external surface of the vehicle. The details of the exclusive use limits are given in 10 CFR 71.47 (a) and (b).

All DOE weapon component and special assembly packages are shipped in exclusive use DOE conveyances. However, the DOE policy of as low as reasonably achievable (ALARA) (see 10 CFR 20) for external package dose rates can be interpreted so as to not allow the higher NRC exclusive use limit, and all DOE weapon component and special assembly packages to date have complied with the nonexciusive limits as outlined in the preceding three paragraphs. Most probably, the exclusive use limit of 1000 mr/hr would exceed the local plant radiation limits where the packages are assembled before shipment and unpacked after shipment.

The ALARA requirement can also be examined with respect to the nonexclusive use dose rate limits. The design of most packages for the shipment of weapon components and special assemblies, such as that shown generically in Fig. 5.1, is usually dictated by considerations from structural, thermal, containment, and criticality aspects. Generally, no specific shielding materials are included in the package, but the external dose rates will in most cases be much lower than the nonexclusive use dose rate limits. The dose rates could conceivably be reduced to or near background levels by including appropriate liners of lead, steel, or other shielding materials into the package designs. However, this package enhancement might be considered unreasonable considering the increased package weight and the increased cost of the package fabrication and shipping procedure.

The ALARA concept takes on greater importance if the shipping packages or the inner containers are also to be used for storage and/or several packages are clustered in the same general location. Now, the combined doses from all of the packages, each of which meets the transportation regulatory dose rate limits individually, may pose a radiation hazard.

The 10 CFR 71.47 requirements state only that the dose rates on the conveyance exterior must not exceed 200 mrem/h; they do not address the combined package dose rates in the conveyance interior. The possible use of shipping packages for storage must be examined to determine if additional shielding (over that for transportation alone) is needed for the storage facility to be in compliance with 10 CFR 20.

5.4.5 Safety Analysis Report for Packaging (SARP)

Under DOE Order 5610.12, a detailed document explaining the design, construction, and operation of a packaging together with its contents (a package) must be approved and accepted by DOE before a transportation certificate authorizing the shipment of the package can be issued. It is recommended that this document, a Safety Analysis Report for Packaging (SARP), be prepared in general conformance with NRC Regulatory Guide 7.9.[4]

This NRC guide has been prepared as an aid in the preparation of applications to NRC for approval of packaging to be used for the shipment of Type B and fissile radioactive material in accordance with 10 CFR 71. In addition, a Packaging Review Guide[5] has been prepared by DOE to maintain the quality and uniformity of reviews of SARPs which are submitted to DOE for package certification.

5.4.6 Calculational Analysis

Following the recommendations of the NRC Guide 7.9, Sect. 5 of a SARP for a particular packaging and contents will contain a calculational analysis confirming that the external dose rates for the package are in compliance with the limits as specified in 10 CFR 71.47 and 10 CFR 71.51. Following the recommendations of the DOE Review Guide, the confirmation of compliance with the regulations is acceptable if the calculations follow from commonly accepted radiation shielding analysis practices.

5.4.7 Measurements and Calculation Comparisons

None of the applicable regulations, orders, or guides require or recommend that experimental measurements of the external package dose rates be made or reported in a SARP. Any package loaded, or ready for loading, on or in a conveyance for transport will be measured for external radioactivity as part of the local plant health physics and radiation monitoring program. If any of the package dose rates are found to exceed the regulatory limits at this time, the package can not be shipped. The primary reason for the shielding analysis is to ensure beforehand that these limits will not be exceeded. The measurements will record the dose rates resulting from the radioactive source material at the time of loading. Whereas, the analysis must account for any source buildup or decay that may occur during the period for the authorization certificate and compute dose rates based on the maximum possible source strength during that time. The use of the transportation package for long-term storage of its contents will increase the time period for source maximization. The conservatisms built into the source calculation and other modeling items will in general give calculated dose rates that exceed the measure values, sometimes by substantial amounts. If the measured values are greater than the calculated values, all measurements and calculations should be carefully examined to explain this difference, even when none of the regulatory dose rate limits are exceeded by either method. In some cases for weak radioactive sources, the external package dose rates may be comparable to or less than the local background radiation, especially for the off-surface dose rates. For unirradiated uranium source material, the package surface neutron dose rates will ordinarily be comparable to background levels.

In addition to providing a comparison of independent analytic methods, measured dose rates present a means of comparison for determining the validity of analytic results. Calculated dose rates from a detailed analytic model of the shipping package should compare with the measured values to within the limits of uncertainty for both the analytic and experimental methods. Comparison of experimental and calculated results should be a part of any ongoing packaging and SARP shielding analysis program to interpret and analyze the degree of conservatism built into the calculation model. Little or no conservatism in the shielding model may result in unnecessary time and expense for both personnel and computer charges. Overly conservative shielding models that produce calculated dose rates approaching the regulatory limits should be avoided. These large values might easily be misinterpreted as actual values, which would not well serve a packaging program or the nuclear industry in general. However, simple and conservative models that produce dose rates well below the limits provide a convenient means of economically satisfying the applicable regulatory codes, orders, and guides. If the ALARA requirement is to be invoked (see Subsect. 5.4.4) and/or the package is to be used for storage together with other similar packages, attempting to compute actual external package dose rates with little or no conservatisms may be necessary. In this manner, a realistic estimate of any additional package shielding necessary to meet the ALARA requirements or storage facility radiation limits can be determined.

The measurements of the package dose rates will ordinarily be done with a hand-held survey meter used to monitor radiation levels in the plant where the package is assembled. Although these meters go through required periodic calibration procedures, the dose rate readings cannot be as accurate as those from more scientific, fixed radiation-detector equipment used in a laboratory environment. Any inaccuracies will be exaggerated for hand-held, off-surface package dose rate readings.

Each DOE installation in which radioactive materials are made packed, shipped, unloaded, or stored will have local health physics procedures for radiation monitoring. The use and calibration of detection equipment will be done according to ANSI N323-1978.[6] All packages being loaded or unloaded will be surveyed to ensure that any regulatory dose rate limits are not exceeded. Some plant procedures may not require that packages be lifted so that bottom dose rates can be measured. For many packages, however, the maximum exterior doses are on the bottom surface. Also, most procedures do not require that inner containers be surveyed before they are placed into the package. Measurements off the surface of the inner container would give an indication of the values calculated for the HAC mode.

In the past, measurements alone were sometimes used as the means of satisfying the regulatory limits for shipping packages. However, problems arose in the SARP review and certification process since the measurements reported in SARP preceded the actual shipment time by several months, or even years, and it was difficult to establish that the measured dose rates reported in the SARP corresponded to the maximum values for any applicable packages to be shipped at a later time. It may still be possible to pursue a measurement-only program to satisfy the package dose rate regulations, but as outlined in Subsect. 5.4.5, it is recommended by NRC Regulatory Guide 7.9 that a shielding analysis also be performed.

5.5 SHIELDING ANALYSIS

In general, the primary purpose for the shielding analysis for a package designed to ship weapon components or special assemblies is to establish in Sect. 5 of the applicable SARP that all regulatory requirements concerning the package exterior dose rates have been met. It is unlikely that a package designed to meet the other requirements pertaining to structural integrity, containment, thermal limits, criticality, etc., would exceed the exterior dose rate limits. The primary reasons for this situation are:

1. The radioactive source contains only unirradiated material, the exception being irradiated material in concentrations measured in ppm or less relative to the unirradiated material.

2. The mass of radioactive source material is limited by conservative criticality safety analysis requirements, and usually to a lesser extent, thermal and material containment requirements.

3. There are no radiation streaming paths, penetrations, voids, or other package geometric irregularities which would permit unattenuated radiation to pass directly from the source material to the package exterior (streaming and scattered radiation along streaming paths is often the primary shielding concern in reactor shields, spent-fuel shipping casks, high level radiation waste depositories, etc.).

4. The stainless steel inner container, other package materials, and source material self-shielding will sufficiently attenuate the largest portion of the radioactive source, the low-energy gamma rays.

In general shielding analysis for determining the dose resulting from a proposed design or for comparing results from measurements, the shielding model is constructed as close as practically possible to the design model or experimental setup. Homogenization or omission of materials for conservatism is usually not an issue if a realistic dose or dose rate is to be calculated. In mixed-field radiation of neutrons and gamma rays, neutron source material can act as a gamma ray shield and neutron shield material can act as a gamma ray source. Because of these and other considerations, shielding analyses are most often done using nominal or average values for material densities and dimensions. Such would be the case for long-term storage or ALARA analysis.

However, a shielding analysis for a SARP for authorization only to ship radioactive material such that the external package radiation levels do not exceed regulatory limits can be achieved with a degree of conservatism similar to that for a criticality safety analysis. It is necessary to show only that conservatively calculated dose rates, such as $k._{1}$ for criticality, will not exceed some limiting value. These conservatively calculated dose rates should always exceed measured values, sometimes by substantial amounts.

Some of the items to be considered in a conservative shielding analysis are:

1. Compute the radioactive source strength at or very near the time of maximum strength. This task can be done with a few calculations to determine the time within a year or a half year.

2. Include any high radiation-producing isotopes, such as trace amounts of ^{236}Pu in plutonium or ^{232}U in uranium, at the highest measured or theoretical values.

3. Include any trace element impurities in the source material, such as B, Be, F, etc., at maximum measured values to maximize the (α, n) reaction neutron source.

4. An even more conservative (α, n) neutron source is to use data in the source generation code for a uranium oxide or water medium with the source material. This method could overpredict the neutron source by more than an order of magnitude.

5. Use the maximum measured or theoretical density for the radioactive material in computing the total source strength or source normalization factor.

6. In the calculation of the dose rates, after the source has been determined, use the minimum measured values of density and dimensions for shielding materials and the maximum values for materials that produce sources during the dose rate calculation [e.g., secondary gamma rays, induced fission neutrons, (n, 2n) neutrons, etc]. The concerns regarding the maximum and minimum values of material density and dimensions may lead to conflicting choices due to the production of secondary radiation, and use of nominal or average values may be necessary for dose rate calculations. The effect on the calculated dose rates from the use, or lack of use, of

maximum or minimum measured densities and dimensions should be at most only a few percent. In contrast to the effect of such variations on the results of a criticality safety analysis, small-percentage changes in the results of a shielding analysis are usually of little concern.

7. The overall efficiency of a shielding calculation may be enhanced by conservatively omitting from the calculational model nonsource materials that present some geometric, compositional, or other difficulty. This difficulty may appear as a result of some complication in the weapon component or special assembly contents or in the packaging material. Uranium containing none of the ^{232}U isotope may be conservatively omitted from a shielding model when the computed exterior package dose rates result primarily from gamma rays in other radioactive source material. The gamma ray shielding properties of uranium outweigh any contributions to the dose rate from its small neutron and gamma ray source when no ^{232}U is present.

8. The overall efficiency of a shielding analysis will also be increased by conservatively omitting the outer packing/insulation material and the shipping drum. The NCT and HAC shielding models are identical, except for detector locations, and all dose rate calculations from both configurations can be made simultaneously. For most shipping packages containing weapon components or special assemblies, the omission of this outer packaging material will increase the calculated exterior dose for NCT by 50% at most.

9. If a series of packages is to be shipped containing weapons components or special components that are similar in some way from package to package, the dose rates computed for the most conservative of the models may be used to represent all the packages. A situation often encountered is the necessity to ship many radioactive source material parts separately (or in small groups of two or three each in a package) in many packages with identical packaging materials.

If all these parts are identical (or nearly so) in composition and differ only in mass and shape, one dose rate calculation for the most massive part (or parts) in the most conservative position in the inner container can be used for all packages. It may be convenient to change the part shape to some generic shape spread out over some portion of the top, bottom, and side of the inner container with no internal packing material for three separate calculations to give maximum possible dose rates for presentation in Table 5.1 of SARP. The procedure for using conservative generic shapes and masses may also allow the reduction or omission of any security classification in connection with the shielding analysis and presentation. (The methods proposed in this item may grossly over-predict the actual package dose rates.)

10. The neutron source dose rate calculation will generally be much more expensive than the corresponding gamma-ray source calculation, even when the calculated value is comparable to background levels as for uranium or thorium. Compliance with the required analysis and regulatory limits can be accomplished with a very conservative and approximate method. The neutron dose rate can be computed from the neutron source strength and spectra using a point source in void flux calculation as explained in Sect. 8.5. This method is not applicable to plutonium.

11. Many conservative and approximate methods are available for package source strength and dose rate calculations that are specific to a particular computer code or group of computer codes. These methods will be discussed with the individual codes in the following sections,

The use of conservative and approximate computational methods for determining source strengths and exterior package dose rates can greatly improve the overall efficiency in the calculations, SARP presentation, and review of the SARP shielding section for establishing that all regulatory requirements

Table 5.1. Computer programs used for generation of radiation source terms

Code (RSIC CCC No.)/developer	Reference	Description and comments
EPRI-Cinder (CCC-309) Los Alamos National Laboratory	15	Point-depletion code for computing actinide and fission-product atom densities. Solution via Bateman equations. Auxiliary codes required for generating radiation source spectra and strengths. Other code versions are CINDER2, CINDER3, CINDER7, and CINDER 10. Data libraries and availability vary between versions.
ORIGEN (CCC-217) Chemical Technology Division Oak Ridge National Laboratory	8	Point-isotope generation and depletion code. Solution by matrix exponential method. Actinide, fission-product, and light element libraries available. Photon source spectra (fixed groups) and neutron source strength generated using outdated data and/or analytic functions.
ORIGEN-2 (CCC-371) Chemical Technology Division Oak Ridge National Laboratory	11	Significant updated of the ORIGEN code to remove deficiencies, improve input features, provide new and better data libraries (actinide, fission product, and light elements). Photon source spectra (fixed groups) and neutron source strength improved over ORIGEN code. Well documented and widely used.
ORIGEN-S (CCC-545) Nuclear Engineering Applications Section Oak Ridge National Laboratory	12	Significantly update version of the ORIGEN code developed for the SCALE system. Decay data and photon data same as for ORIGEN2. Radiation source (n and γ) strength and spectra provided in user-specified or default multigroup energy structure. Well documented.
RIBD-II (CCC-79) Pacific Northwest Laboratories	17	A subroutine within the ISOSHLD II and III point-kernel codes. Performs a reactor point-depletion analysis to produce gamma source spectra for fission products. Fission-product data libraries available for generic thermal and fast reactors.

have been met. The use of any of the preceding methods, either singly or in combinations, will affect the degree of overprediction of the dose rates compared with the measurements at the time of shipment.

5.6 COMPUTER CODES AND DATA SETS

The computer codes and data sets to be discussed in this section represent only a small number of those available. No attempt will be made to rank the codes in any order according to theoretical exactness, ease of use, availability of necessary data, etc. In practice, the decision to use a particular code is often based on such items as its availability on a convenient computer operating system and the close proximity of personnel with experience in its use. The Radiation Shielding Information Center (RSIC) at the Oak Ridge National Laboratory is the best general source for codes and advice on their use in the preparation of SARP shielding analyses for weapon component and special assembly shipping packages.

5.6.1 Codes for Radiation Source Generation

Table 5.1 provides a list of the codes that appear suitable and/or are commonly used for generating radiation source terms. All the codes are available from the RSIC computer code collection. The information in this and the next section is based on two earlier compilations of computer code use for radiation shielding applications.[7]

The most widely known code in the table is the original ORIGEN[8] code that serves as a basis for several of the other codes: ORIGEN-JR,[9] KORIGEN,[10] ORIGEN2,[11] and ORIGEN-S.[12] Although ORIGEN is still widely used, the four updated codes provide significant improvements over the original version. These improvements are well documented in the respective references for each updated code. Of the four updated ORIGEN codes, the U. S. codes ORIGEN2 and ORIGEN-S are the most widely used.

The ORIGEN2 code is the most popular of the updated versions of the ORIGEN codes. The library data and radiation source-term evaluation offer a significant improvement to the ORIGEN code. ORIGEN2 provides the gamma spectra in an 18-energy-group format that matches the group format of the 22n-21 γFCXSEC[13] cross-section library for all but the last few high-energy groups. However, ORIGEN2 provides only the neutron source strength. Thus, the analyst must generate a neutron spectrum in the required group structure when using ORIGEN2. Also, if using a gamma cross-section library with a group structure other than that for which the source is provided, the analyst must adjust or interpolate the ORIGEN2 gamma source spectrum. ORIGEN2 is relatively easy to use and has several built-in data libraries for typical use.

The ORIGEN-S code provides complete neutron and gamma source spectra in any multi-energy group format. Thus, the shielding analyst is provided with the flexibility to select a multigroup cross-section library without needing to interpolate from one fixed group to another. ORIGEN-S outputs the separate spectrum for (α, n) and spontaneous fission neutrons and the total neutron spectrum. As a result of this flexibility, the ORIGEN-S input is more complex than that of ORIGEN2.

The CINDER series of codes represents the major alternative to the ORIGEN codes in the United States. As with the ORIGEN-type codes, several updated versions of the CINDER[14] code exist that have been developed and are now in use at Los Alamos National Laboratory (LANL). Of these codes, EPRI-CINDER[15] is the only one that is publicly available from RSIC. The neutron spectrum may be produced by using the SOURCES code.[16]

Although older and far more limited than the ORIGEN- or CINDER-type codes, the RIBD-II code[17] has been widely used for spent fuel gamma sources because it is interfaced with the point-kernel code ISOSHLD[18] to provide an easy-to-use procedure for gamma-ray source generation and shielding analysis. The RIBD routine is limited to evaluating the gamma source spectra from only fission products and requires another routine call BREMRAD[19] to evaluate the Bremsstrahlung source spectra.

In a more complex yet more complete fashion, the 5A52 shielding sequence of SCALE[20] uses ORIGEN-S to generate radiation source spectra for subsequent input to a radiation transport module.

5.6.2 Codes for Radiation Dose Evaluation

This section will provide a discussion of relevant codes that use the three basic techniques-point-kernel codes, discrete ordinates codes, and Monte Carlo codes-and the three basic geometric models-one-, two-, and three-dimensional (1-D, 2-D, and 3-D).

5.6.2.1 Point-kernel codes

Point-kernel codes provide approximate, conservative evaluations of the primary gamma-ray dose from a source. Moreover, these codes are inexpensive, computationally fast, and far less cumbersome or complex relative to discrete ordinates or Monte Carlo codes.

Table 5.2 provides a list of three of the more popular point-kernel codes. Of these, the QAD family of codes has enjoyed the greatest popularity and use. Originally developed at LANL in the 1960s, it has been successfully updated by a variety of users. The latest and best version of QAD is QAD-CGGP,[21] which features the flexible, 3-D combinatorial geometry (CG) package, the standard build-up

Table 5.2. Computer programs based on point-kernel techniques used for radiation dose evaluation

Code (RSIC CCC No.)/developer	Reference	Description and comments
ISOSHLD (CCC-7) Pacific Northwest Laboratories	18, 23	Code uses direct point-kernel techniques to generate gamma doses for common geometric models. Source may be input directly or calculated via the RIBD routine. An extremely user-friendly, for-sale version called MICROSHIELD is available for the PC.
PATH GA Technologies	24	Proprietary point-kernel code available from GA. Buildup factor specification can vary with source and dose point. Claims to be "fully validated and suitable for nuclear licensing applications."
QAD (CCC-48, 307, 346, 396, 401, 448, 493) Original code from Los Alamos National Laboratory	21, 22	The QAD family of codes makes up seven different code packages in the RSIC code collection. All use direct point-kernel techniques and differ principally in the available geometry package, source integration method, build-up factor, interpolation scheme, and ease of input. Latest buildup factor data and attractive geometry in new QAD-CGGP (CCC-493). Most widely used of point-kernel codes.

factor data of ANS-6.4.3,[22] and the geometric progression (GP) fitting function for the build-up factor data. The latter two features represent a substantial improvement over the basic build-up factor data and interpolation scheme now used in most other codes. The improvement is most evident for shield materials of low- or very high-Z number and/or low-energy (< 0.5 MeV) photons. The CG geometry feature of QAD-CG and QAD-CGGP is attractive because the geometry input description can be easily interchanged for use in combinatorial geometry versions of MORSE (see the paragraph on Monte Carlo codes in this section).

The ISOSHLD[18] code has the capability of generating an irradiated fission-product source using the RIBD routine. Also, an extremely user-friendly version of ISOSHLD (called MICROSHIELD[23]) has been developed in a proprietary package. The other codes in the table also have attractive but less noticeable features that distinguish them from QAD-CGGP in terms of the assessment criteria. One might hypothesize that proprietary for-sale codes such as MICROSHIELD or PATH,[24] which see their users as valuable for-profit customers, may have a more systematic approach to quality assurance criteria maintenance, ease of use, and validation.

Note that the new build-up factor data and GP fitting functions could be added rather easily to any of the point-kernel codes mentioned in this section. Many, if not all, of these codes will probably be updated at some time once the advantage of the new data and fitting function is realized.

5.6.2.2 Discrete ordinates codes

The discrete ordinates codes provide a numerical solution to the Boltzman transport equation and, as such, are more appropriate for general applications than are point-kernel or other approximate codes.

However, the added complexity of these codes requires greater computational resources and user involvement.

Table 5.3 lists the premiere discrete ordinates transport codes and known auxiliary codes that facilitate accurate radiation dose evaluations. The table includes three 1-D discrete ordinates codes (ANISN, ONEDANT, and XSDRNPM), two 2-D discrete ordinates codes (DOT series and TWODANT), and three auxiliary codes for use in evaluating doses at point detectors. Geometry requirements and/or level of desired computational effort typically dictate the selection of a 1-D or a 2-D code. Sometimes a shield configuration can be reasonably approximated in one dimension (plane, cylinder, or sphere), and the 1-D programs can combine the accuracy of discrete ordinates with the near speed of point-kernel techniques.

The 1-D ANISN code[25] in Table 5.3 is probably the most widely used radiation shielding code (point-kernel or otherwise). Using the numerical solution technlques of ANISN, the XSDRN code[26] evolved from its initial release to the version called XSDRNPM-S[27] with the following added features: 1) solutions using double-precision flux arrays to circumvent potential convergence difficulties; 2) more user-friendly input (availability of parameter default values, automatic generation of appropriate angular quadrature, etc.); 3) increased flexibility in the input/output and processing of multigroup cross-section data; and 4) inclusion within a well maintained modular code system called SCALE.[28]

The ONEDANT code[29] is the third 1-D code noted in Table 5.3. The code is much newer than ANISN, XSDRNPM, and the older LANL code called ONETRAN.[30]

For problems requiring 2-D discrete ordinates shielding analyses, the DOT code series has become the international standard. The latest version, DORT,[31] represents a significant advancement

Table 5.3. Computer programs based on discrete ordinates techniques used for radiation dose evaluation

Code (RSIC CCC No.)/developer	Reference	Description and comments
ANISN (CCC-82,253-255, 314, 514) Oak Ridge National Laboratory	25	General 1-D discrete ordinates coupled neutron-gamma radiation transport code. Most popular version is ANISN-ORNL (CCC-254). Flux or activities at a detector site can be evaluated. The Westinghouse version, ANISN-W (CCC-255), and a recent version for EG & G (CCC-514) are available for an IBM PC.
XSDRNPM (CCC-545) Oak Ridge National Laboratory	27	1-D discrete ordinates coupled neutron/gamma ray transport code based on ANISN. Extends ANISN capabilities to provide user-friendly features, automatic quadrature generation, and flexibility in weighting cross sections. Easily coupled to XSDOSE for doses exterior to shield.
ONEDANT (CCC-547) Los Alamos National Laboratory	29	General 1-D discrete ordinates coupled neutron-gamma radiation transport code. Modular program developed to be very user friendly. Fluxes and/or activities provided at detector points.
DORT (CCC-543) Oak Ridge National Laboratory	31	General 2-D discrete ordinates coupled neutron-gamma radiation transport code. Earlier versions are obsolete. DOMINO II (PSR-162) couples DOT IV to the Monte Carlo MORSE-CG code (CCC-203). Fluxes and activities calculated. Excellent documentation of theory and techniques.
TWODANT (CCC-547) Los Alamos National Laboratory	30	General 2-D discrete ordinates coupled neutron-gamma radiation transport code. TWODANT is basically the ONEDANT package with the 1-D SOLVER module replaced with a 2-D SOLVER module.
SPACETRAN (CCC-120) Oak Ridge National Laboratory	35	Evaluates dose for detectors at various distances from the surface of a cylinder. Integrates ANISN leakage or DOT 3.5 flux data. Eliminates potential ray effects in air or void outside a cylinder. Not accurate for detector points near the cylindrical surface.
FALSTF (CCC-351) Oak Ridge National Laboratory	34	Calculates doses exterior to a shield based on DOT 3.5 calculated fluxes in cylindrical geometry. Doses evaluated as sum of last flight contributions from shield regions. Eliminates potential ray effects in air or void outside cylinder. Only available for DOT 3.5.
XSDOSE (CCC-545) Oak Ridge National Laboratory	36	Used in conjunction with XSDRNPM (or ANISN) to compute the neutron/photon flux and resulting dose rate at various points outside a finite cylinder, sphere, rectangular slab, or circular disc. Uses direct line-of-sight integration of surface angular flux over the surface. Eliminates potential ray effects from discrete ordinates outside shield. Extremely easy to use.

in computing efficiency and speed; however, most problems of reasonable size still require substantial computer resources. The DOT codes were developed primarily for radiation shielding analysis, whereas TWOTRAN and TWODANT were developed in a reactor/core physics environment. This difference in emphasis explains why DORT is typically selected where shielding calculations are of prime importance. The DORT manual provides an excellent explanation of the basic theory and numerical techniques employed in the code.

Although they are not included in Table 5.3, a few 3-D discrete ordinates codes such as TORT[32] and THREETRAN[33] are available. These codes are practical only on vector operation computers such as CRAY.

Table 5.3 includes three auxiliary codes that were developed to provide an easy means of accurately evaluating the flux or dose at a point exterior to a shield. For problems in which doses are required exterior to a shield in a low scattering medium (air, void, etc.), extension of the discrete ordinates spatial mesh into the exterior medium is often unattractive for the following reasons:

1. A penalty is paid for the extra spatial mesh (typically a fine spatial mesh and angular quadrature are needed for curvilinear geometries).

2. For 1-D problems, there is no good way of accounting for the finite dimensions of the shield from which the radiation leaks.

3. Ray effects in multidimensional problems are very difficult to alleviate and can yield unreliable results.

To alleviate these problems, FALSTF,[34] SPACETRAN,[35] and XSDOSE[36] were written for use with the ANISN, DOT, and XSDRNPM codes. Although they are available for DOT 3.5, public versions of FALSTF and SPACETRAN are not available for the DORT code. The SPACETRAN code is computationally more efficient than FALSTF but is inaccurate close to the shield and can be unreliable if an inappropriate spatial mesh or angular quadrature is used. The XSDOSE code has the best numerical techniques used to eliminate the difficulties inherent in the 1-D SPACETRAN method.

5.6.2.3 Monte Carlo codes

Table 5.4 lists the two Monte Carlo codes that are in general use for performing radiation shielding analyses. Because of its easy-to-use features, accessible ANISN-formatted cross-section data, and ready availability, the MORSE[37] code is a much-used Monte Carlo code for radiation shielding. The latest versions of MORSE (CGA[38] and SGC/S[39]) use multigroup cross sections, a wide variety of source and particle biasing features, and a CG package with nested array features.[40]

Although the MORSE codes are still widely used, the MCNP code[41] developed at LANL has rapidly gained in popularity. The MCNP code was once regarded as a highly specialized code that was difficult to use, but MCNP developers have made a concerted effort to retain the sophisticated attributes of the code and still provide an easy-to-use and readily acceptable tool. The main areas of sophistication concern the use of point-energy cross-section data (supplied with the code) and development of "automatic" biasing schemes. The automatic biasing schemes are an attempt to reduce the required user expertise in analyzing a problem. The MCNP code represents the current technology in Monte Carlo code development for radiation shielding.

Table 5.4. Computer programs based on Monte Carlo techniques used for radiation dose evaluation

Code (RSIC CCC No.)/developer	Reference	Description and comments
MCNP (CCC-200) Los Alamos National Laboratory	41	General-purpose Monte Carlo code for coupled neutron/photon particle transport. Capable of handling point energy and discretized cross-section data. New features for automatic generation of importance functions. Flexible geometry capabilities. Source and response estimator specification flexible and easy to use. Well-supported program with constant improvements and updates. Well documented.
MORSE (CCC-127, 203, 258, 261, 277, 368, 394, 545, 471, 474) Oak Ridge National Laboratory	37, 38, 39, 41	General-purpose multigroup Monte Carlo code for coupled neutron/photon particle transport. Latest versions from ORNL in CCC-345 and CCC-474 have a popular, easy-to-use geometry package capable of generating multiple arrays within array

There are other state-of-the-art Monte Carlo codes in use that are not included in Table 5.4. Many of these codes are either unavailable from RSIC or available in either incomplete or out-of-date versions. The list includes SAM,[42] TRIPOLI,[43] TART,[44] and COG,[45] The TRIPOLI code has one of the most sophisticated biasing techniques of any existing Monte Carlo code.

5.6.3 Cross-section Data Libraries

This section provides a review of the various types of available cross-section libraries and discusses those now being used. The primary differences between various types of cross-section sets are outlined. More so than with radiation transport codes, the "best" multigroup data library will probably vary from application to application. Keeping in mind the general assessment criteria, this section notes libraries that have been widely used and those that need further assessment.

Broad-group libraries have traditionally been developed for production use and, whether generated from a fine-group library or developed directly from evaluated data, are typically application-dependent libraries. Some of the older broad-group libraries generated directly into a discrete ordinates format are shown in Table 5.5. Typically, these libraries were developed and used successfully for a given project, and results obtained with the data were published. The first library, CASK,[46] was developed for depleted uranium shipping casks with a water-filled cavity. The energy grouping was done based on typical spent fuel spectra. The data source for this library is quite old, and the resonance correction for ^{238}U is inadequate if the subcritical multiplication source is important to the dose. However, CASK has been one of the most widely used (for all applications) ANISN-formatted libraries. The second library, FEWG1,[47] was developed for radiation transport through concrete and air. The work was sponsored by the Defense Nuclear Agency (DNA), and the group structure was developed for applications with source spectra from nuclear weapons. The library has an extensive selection of kerma

Table 5.5. Some broad-group libraries in discrete ordinates format developed for specific applications

Library	Contributor	Energy groups	Processor	Source	Application (No. elements)
DLC-23/ CASK	ORNL	22n, 18g	SUPERTOG, POPOP4 (Collapse)	ENDF/B-II POPLIB	Shipping Casks/(29)
DLC-31/ FEWG1	ORNL/ DNA	37n, 21g	AMPX	ENDF/B-IV, DNALIB	Concrete, Air (43)
DLC-36/ CLAW-IV	LANL	30n, 12g	NJOY	ENDF/B-IV	General/(48)

response functions for various materials. The CLAW-IV[48] library was developed for shielding analyses related to weapons applications.

A more attractive procedure for developing a coupled broad-group library has been to process (resonance shielding and temperature correction) and collapse a fine-group, pseudoapplication-independent library to create a production, application-dependent library. A collection of these libraries is shown in Table 5.6, all of which are ANISN formatted. The BUGLE-80[49] and SAILOR[50] libraries are nearly identical.

The BUGLE-80 library and its parent, VITAMIN-C, are listed in ANSI/ANS-6. 1.2/1983 as suitable cross-section sets for nuclear radiation protection calculations. The standard lists the processing procedures and verification efforts required to be included in the standard. Testing of the BUGLE-80 library was done primarily for concrete shields, and resonance processing was not done on nonconcrete nuclides. The weighting spectrum used in the collapse from VITAMIN-C was that of a concrete medium. Thus, the validity of the library for nonconcrete-shielded applications needs further testing.

The FXSEC library was developed for fuel cycle shielding analyses. A generic fusion-fission-1/E-Maxwellian spectrum was used to collapse from the VITAMIN-C group structure. Resonance self-shielding was performed for three background cross sections (composition dependent)—0. 1, 1000, and 10^8 b/atom. Macroscopic cross sections are available with appropriate resonance processing for several mixtures.

As stated in the preceding paragraphs, the ANISN format, or more generally, the discrete ordinates format, is a “working” format; that is, the radiation transport codes read these formats directly. No further resonance or temperature correction is possible. A new approach generates a broad-group

Table 5.6. Some broad-group application-dependent libraries developed from fine-group libraries in "flexible" format

Library	Contributor	Energy groups	Master library source	Application (No. elements)
DLC-75/ BUGLE-80	ORNL, ANS-6.1.2	47n, 20g	DLC-41/VITAMIN-C	Standard for concrete, LWR shielding/(66)
DLC-76/ SAILOR	SAI, ORNL	47n, 20g	DLC-41/VITAMIN-C	BWR and PWR radiation transport analysis/(58)
DLC-85/ FCXSEC	ORNL	22n, 21g	DLC-41/VITAMIN-C	Fuel cycle shielding analyses/(Many)

library with a selected weighting spectrum but retains the "flexible" AMPX-like format that provides the neutron resonance information. The shielding libraries provided with the SCALE system were the first to use this approach. The libraries are provided in the AMPX "master" format, and the SCALE sequences use BONAMI and NITAWL modules to do the resonance and temperature correction (cheap, relative to the radiation transport analysis) for each particular problem and alter the format from a master to an AMPX "working" format. Libraries in SCALE with no resonance information (e.g., CASK 22n-18γ) follow the same procedure, but no actual processing takes place; that is, BONAMI and NITAWL are used merely to change the master format to a working format. Installations that do criticality analyses have resonance processing codes available because resonance processing is of extreme importance in evaluating an accurate neutron multiplication factor.

To date, very few libraries follow the preceding approach. Of those that do, the 27n-1 8γ library in SCALE is the most prominent. The neutron data were collapsed from the CSRL library, and the gamma data were created directly using various AMPX modules. The library group structure and weighting function were selected to be appropriate for spent fuel shielding applications. The large number (13) of thermal neutron groups can increase the cost of a discrete ordinates shielding analysis unless the outer iterations are limited by code input.

The only other broad-group libraries available in the flexible format are the MATXS libraries from LANL (see Table 5.7). However, the MATXS1 and MATXSS libraries contain nonresonance information and are simply MATXS-formatted versions of CLAW-IV. The only library of potential interest is the CLD-1 16/MATXS6 library. However, too many neutron groups appear to exist for it to be used for production work.

Table 5.7 Some libraries in "MATXS" format

Library	Contributor	Energy groups	Processor (format)	Source	Application (No. elements)
DLC-114/ MATXS1	LANL	30n, 12g	NJOY-II (MATXS)	ENDF/B-II LANL	MATXS equivalent of DLC-36/CLAW-IV/(64)
DLC-115/ MATXS5	LANL	30n, 12g	NJOY-II (MATXS)	ENDF/B-V, LANL	ENDF/B-V equivalent of MATSX2/(87)
DLC-116/ MATXS6	LANL	80n, 24g	NJOY-II (MATXS)	ENDF/B-V	Fast reactor shielding, fusion/(91)
DLC-117/ MATXS7	LANL	69N	NJOY-II (MATXS)	ENDF/B-V	EPRI-CPM group structure, PWR studies/(80)

5.6.4 The Scale Computational System

As evidenced in earlier sections, the number of different techniques, codes, and data libraries can confuse even a routine user as to the appropriate procedure for obtaining an accurate dose evaluation. This situation, combined with the expertise required to use (and not abuse) many of the available analytic tools, forces many users to employ the tools with which they have had the most experience (or that are the easiest to use), whether each is the best tool for a particular problem. This section provides a summary of a modular code system called SCALE that was developed in an effort to ease many of the burdensome input and code interface requirements necessary to perform a complete shielding analysis for a specific category of applications.

The SCALE system was developed to be an easy-to-use analytic tool for performing criticality, shielding, and heat-transfer analysis of nuclear facilities and packages. As a modular code system, SCALE is designed to provide common data interface files, input format, and data processing procedures for system analysis. The development concept was: 1) use well-established computer codes and data libraries, 2) have an easy-to-use input format designed for the occasional user and/or novice, 3) combine and automate analyses requiring multiple computer codes or calculations into standard analytic sequences, and 4) be well documented and publicly available.

A host of validated data bases, (e.g.,material compositions, thermal properties, and cross sections)were also incorporated to allow easy input (via key words) and data accessibility. The analytical sequences are automated to perform the necessary data processing (e.g., problem-dependent resonance self-shielding and temperature correction of cross sections), generate the input to well-established computer programs (functional modules), initiate module execution in proper sequence, and perform any needed post-processing of the analytic results. Thus, the user is able to select an analytic sequence characterized by the type of analysis (criticality, shielding, or heat transfer) to be performed and the geometric complexity of the system being analyzed. The user then prepares a single set of input for the control module corresponding to this analytical sequence. The control module input is in terms of easily visualized engineering parameters specified in a simplified, free-form format. The control modules use this information to derive additional parameters and prepare the input for each of the functional modules in the analytical sequence. Back-to-back execution of individual modules is allowed.

The shielding analysis capabilities developed for the SCALE system focus on many of the well-established codes and libraries. Radiation transport is performed by the 1-D discrete ordinates code XSDRNPM and the multidimensional Monte Carlo code MORSE-SGC, which uses the MARS combinatorial geometry package for easy modeling of complex geometries. (Multidimensional discrete ordinates codes were omitted because of geometric restrictions and difficulty with incorporating them in an automated sequence.) These radiation transport codes and other SCALE modules for cross-section processing (BONAMI, NITAWL), source generation (ORIGEN-S), and dose evaluation (XSDOSE) are incorporated into three shielding analysis sequences—SAS1, SAS2, and SAS4.

SAS1 is basically a user-friendly tool for cross-section preparation and subsequent 1-D shielding analysis using XSDRNPM-S and XSDOSE. SAS2 automates all the steps of a complete shielding analysis: 1) a depletion and decay analysis for a specified assembly geometry and irradiation history, 2) generation of gamma and neutron source strength and spectra, and 3) a 1-D radial shielding calculation (XSDRNPM-S) and dose evaluation (XSDOSE) for a transport/storage package.

SAS4 is designed to eliminate user interaction in selecting Monte Carlo biasing parameters for deep-penetration shielding problems. All of the required biasing parameters are derived from results of an adjoint XSDRNPM-S calculation and automatically input to MORSE so that the user is rid of this difficult input task. A simplified input option is also allowed for some geometry models. Of significance is the fact that homogenous and heterogeneous spent fuel models are easily specified. This type of application-specific simplification makes the use of the complex, general-purpose radiation transport codes easy to use appropriately.

A number of improvements could be made to the SCALE system. However, taken together, the shielding sequences provided in SCALE offer an excellent example of a user-oriented computational tool that can be used for source generation, preliminary shield design, final safety analyses, and review calculations.

5.6.5 Flux-to-dose Conversion Factor (Response Functions)

Normally, the radiation environment is first calculated in terms of particle flux and then translated via response functions to personnel exposure, heat generation, material damage, etc. The response function may be a single conversion factor that is multiplied by the total flux to obtain the total response. More often, the response is a function of energy and is multiplied by the group-wise neutron or gamma-ray flux and summed over groups to yield the total response. In either case, inaccuracies or uncertainties in the response data relate directly to uncertainties in the final answer.

Microscopic response data most applicable to radioactive material transport, storage, and handling are kerma and absorbed dose. These response functions are typically derived from basic ENDF data, and reflect a similar level of accuracy. Some additional uncertainty arises from the exclusion of minor but contributing reactions or from oversimplification of the geometric models used to compute energy deposition or absorbed dose. However, the combined uncertainties of the nuclear data and the approximations used in constructing the response functions appear acceptably small and do not typically require special attention. Kerma responses are available from the International Commission on Radiation Units and Measurements (ICRU) and from the MACKLIB-IB response library. Dose response functions are available from a variety of sources, and are often included in multigroup cross-section libraries available from RSIC. The most commonly used flux-to-dose conversion factors in recent years are those from ANSI/ANS 6.1.1-1977. This ANSI standard provides polynomial coefficients for an analytic fit of the conversion factors as a function of energy. This format allows conversion factors to be easily generated for any selected group format.

The SCALE module DOSE computes neutron and gamma-ray response functions based on the 1977 ANSI standard. Although a newer standard has been released, the author recommends that the older data be used.[51] The new data do not have radiation-type-dependent quality factors included directly in the response functions, as does the older data, and the new data was constructed for computing doses in internal body organs instead of generic surface doses.

5.7 COMPARISON OF CODE CAPABILITIES

5.7.1 Advantages and Disadvantages of Shielding Calculation Codes

5.7.1.1 Point-kernel code

The point-kernel code must be used cautiously. If the shield is made of one solid material and is a simple configuration, the code may give reasonable photon responses. But for shields with several layers of materials and geometric complexities that might yield radiation streaming paths or significant backscatter, at least one discrete ordinates or Monte Carlo calculation should be made to either establish confidence in the point-kernel code or provide correction factors.

The advantages of a point-kernel code are:

1. It is a reliable and inexpensive means for obtaining photon doses in simple systems consisting of a source, solid homogeneous shield materials, and point detector location.

2. It can treat a general 3-D geometry.

The disadvantages of a point-kernel code are:

1. It is restricted to photon transport problems.

2. It is valid only for integrated responses, and energy-dependent results are usually inappropriate.

3. The build-up data for some major materials (such as iron, lead, and concrete) are usually fixed in the codes; for other material for which data are unavailable, an approximation must be made.

4. For shields with several layers of materials, the computed dose can be in great error because the available build-up factor data are usually for individual homogeneous materials.

5. As the build-up factor was usually calculated for an infinite medium, the calculated result by point-kernel code may be overpredicted for a system in which there is no backscatter to the detector, thus increasing lower source energy.

5.7.1.2 Discrete ordinates codes

The advantages of discrete ordinates codes are:

1. The method is deterministic in nature such that errors in calculated results are systematic rather than statistical (as in Monte Carlo approaches).

2. A series of problems having similar characteristics benefit from knowledge of flux densities calculated for a similar case; (i.e., the starting flux guess for the iterative process can be obtained from an earlier calculation of similar problems, leading to faster convergence of the current calculation).

3. Neutrons and photons (including neutron-generated photons) can be treated either simultaneously or separately without any real restrictions.

4. One-dimensional calculations are much faster than similar Monte Carlo calculations, but in two dimensions, the discrete ordinates method has no clear advantage over Monte Carlo in computational speed.

5. Results are obtained throughout the entire system, whereas for Monte Carlo methods reliable results are restricted to only selected portions of the geometry.

The disadvantages of discrete ordinates codes are:

1. The problem geometry must be one of the three basic geometries (rectangular, cylindrical, or spherical) with boundaries placed along coordinate planes, and the importance of the geometry approximations that are required vary with the application and must be either evaluated (via other methods) or rationalized by the user.

2. In multidimensional geometries, the discrete ordinates method can produce nonphysical oscillations in the spatial flux distribution (the so-called ray effect) for radiation transport through void or low scattering media, (the ray effect being primarily a result of localized sources and particle propagation in discrete directions and, therefore, most serious for radiation transport through a void).

3. No basic ground rules exist for defining the best angular quadrature set, space mesh, multigroup structure, and polynomial expansion order for a particular problem. Unfortunately, these user input quantities can be very important to the final dose results.

5.7.1.3 Monte Carlo codes

The advantages of a Monte Carlo code are:

1. It can model complex, 3-D geometries without having to employ approximate techniques.

2. In theory, it provides a convenient means for treating space, energy, and angular dependence continually.

3. Monte Carlo codes such as MCNP can use point-cross-section data and continuous scattering kinematics obtained directly from the evaluated nuclear data files, and thereby the code can eliminate errors in processing multigroup cross-section libraries from the evaluated nuclear data files.

The disadvantages of a Monte Carlo code are:

1. It is a stochastic code and introduces statistical uncertainty in the results.

2. Only selected points, areas, or volumes can be evaluated as opposed to continuous geometric distributions obtained by discrete ordinates codes.

3. Choosing biasing parameters, choosing response estimators, and interpreting the results requires expertise.

5.7.2 Comparison of Code Capabilities

5.7.2.1 Geometry

In general, point-kernel codes and Monte Carlo codes can easily model 3-D geometry, and there is no limitation in developing calculational models. However, discrete ordinates codes are used primarily for 1-D and 2-D geometry, although 3-D discrete ordinates codes have been developed and have undergone extensive verification testing. The 2-D geometry is fixed r-z, x-y, or r-theta only.

5.7.2.2 Buildup factors and cross-section data

Point-kernel codes use dose build-up factors that are determined from experiments and a large number of transport calculations using a flux-to-dose conversion factor, usually from ANSI/ANS 6.1.1-1977. Discrete ordinates codes and the Monte Carlo code MORSE use multigroup cross sections. Inaccuracies in processing group-average values from the evaluated nuclear data files are possible. Appropriate weighting functions, adequate group structure, and proper resonance treatment are major application-specific considerations that are crucial to preparing multigroup cross-sections that give accurate results for a set of applications. The Monte Carlo code MCNP and other general codes can use point cross-sections and are less susceptible to cross section processing inaccuracies.

5.7.2.3 Computational time

Usually point-kernel and 1-D discrete ordinates codes can produce results with a short-computation time. Next in speed are 2-D discrete-ordinates codes. Monte Carlo codes and 3-D discrete ordinates codes require the longest computational times. But if the system is complicated enough to require use of finer energy-group structures, and/or a higher angular quadrature set, and/or finer space meshes, the time difference between discrete ordinates and Monte Carlo codes becomes unclear.

5.7.2.4 Calculated results

Point-kernel codes give reliable integrated responses only. Discrete ordinates and Monte Carlo codes give group-dependent fluxes. The fluxes obtained by a Monte Carlo code have a statistical uncertainty. Empirically, the statistical uncertainty should be less than 10% except for point detectors, and less than 5% when using point detectors, to obtain reliable results.

5.7.2.5 Detector location

Point-kernel codes give results at fixed-point detector locations input by the user. Monte Carlo codes have three kinds of detectors: point, surface, and volume. With respect to the detector locations, the situation is the same with point-kernel codes. Limiting the number and extent of detectors is necessary to obtain results with reasonable statistical uncertainty. Discrete ordinates codes give flux distributions throughout the entire geometry mesh.

5.7.2.6 Limitations

If a calculational model consists of several layers of shield materials, the selection of build-up factors is important to get reliable results by a point-kernel code. A rule of thumb is to use the build-up factor for the final layer if that layer is several mean-free paths thick or to use the build-up factor for the dominant shield layer if the outer layers are only a few mean-free paths thick.

The ray effect is a serious problem in using discrete-ordinates codes. Special techniques must be employed to calculate external detector points correctly. One technique is the last-flight approach, which calculates the flux density at each point detector because of particle scattering from all spacial meshes in the system to each detector (FALSTF). Another technique is to calculate the scalar flux at each detector from the angular flux on the outside surface of the shield (SPACETRAN and XSDOSE). Evaluation of streaming from narrow and long holes or orifices is a difficult task for a discrete ordinates code. Although the holes are correctly expressed by a 2-D model, a higher angular quadrature set or specialized quadrature set must be used to evaluate a streaming component correctly.

The statistical uncertainty associated with Monte Carlo results remains the greatest theoretical limitation to this method. Even when the calculated uncertainty is low (< 10% for shielding and deep-penetration calculations), the results may be unreliable due to an inadequate definition of the code input parameters.

5.8 DESIGN GUIDE RADIATION SAMPLE PROBLEM

The sample problem to be used to illustrate the radiation shielding characteristics of a shipping package will be that of 9 kg of plutonium in a container similar to the Rocky Flats 2030 model. The contents represent an upper limit on the amount of fissile and radioactive material covered in these guides.

5.8.1 Geometry

As illustrated in Fig. *5.2*, the package consists of two identical inner steel containers surrounded by insulation and support material, all enclosed in an outer steel shipping drum. Each inner container holds a 4.5-kg cylinder of plutonium. The following three geometric configurations will be investigated:

1. Each plutonium mass is represented as a thin disk 0.478 cm in height and with a radius equal to that of the inner container inside dimension (12.46 cm). This geometry minimizes the self-shielding in the plutonium, resulting in maximum, or near maximum, possible calculated dose rates exterior to the package. Both NCT and HAC models are calculated for this configuration.

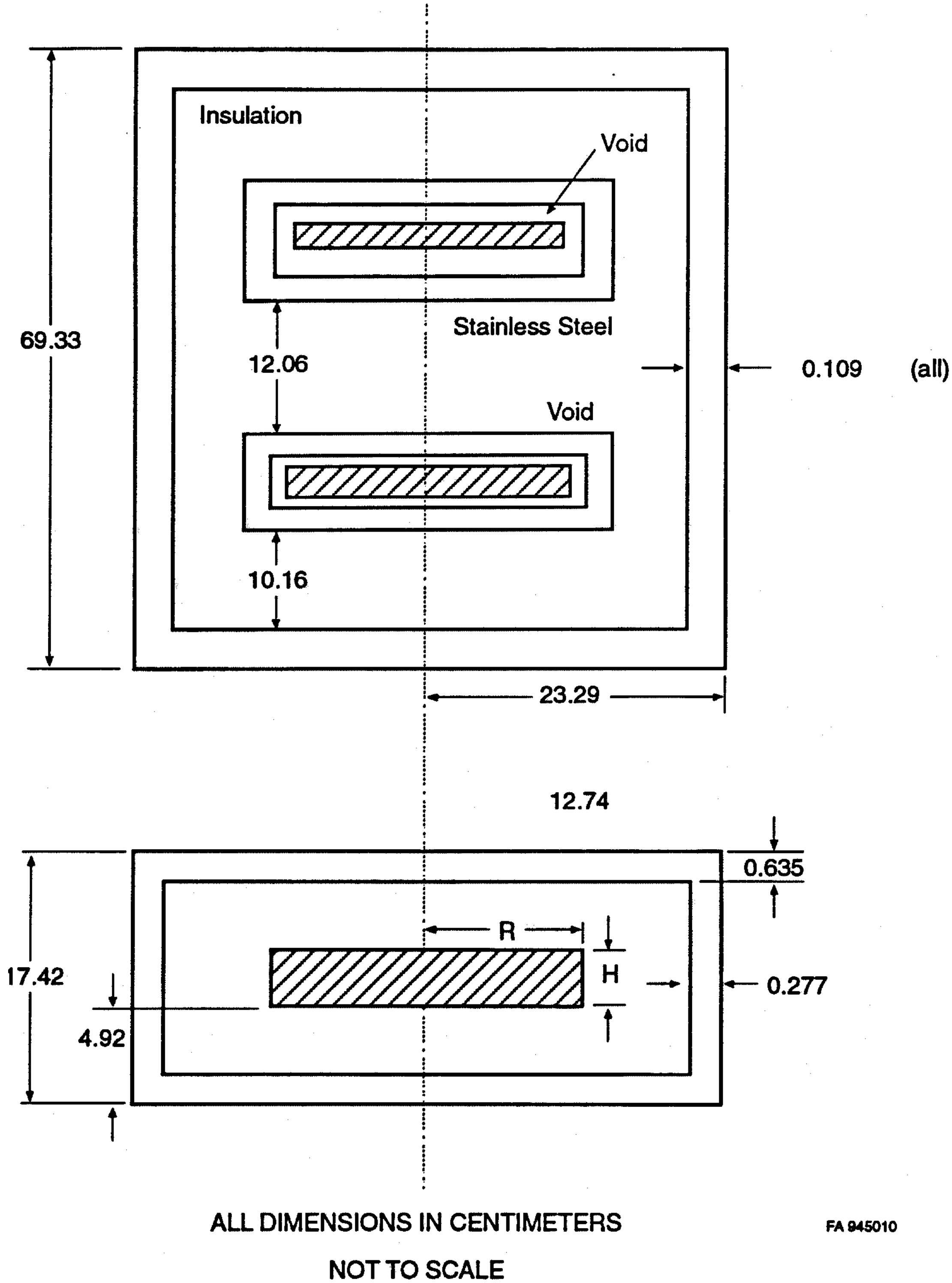

Fig. 5.2. Sample plutonium shipping package.

2. Each plutonium mass is modeled as a "square cylinder" (i.e., the diameter is equal to the height). This geometry gives not only a near maximum possible self-shielding effect but also a near maximum possible induced fission neutron production in the fissile material. Only the NCT mode is investigated here.

3. This configuration is for an isolated inner container, independent of the rest of the shipping package, to be used for long-term storage of the plutonium. The thin disk geometry of configuration number one is used.

All of these configurations model the contents of the inner containers as void, except for the plutonium. In an actual package, the plutonium will be surrounded by some form of wrapping and support material, having little radiation attenuation properties. Because this material may contain voids or other geometric irregularities, its omission facilitates the analytic modeling and also adds a small degree of conservatism to the calculated dose rates.

5.8.2 Neutron and Gamma Ray Source Spectra

The neutron and primary gamma ray source spectra resulting from the radioactive decay of plutonium were calculated with the ORIGEN-S code. The initial isotopic concentrations are shown in Table 5.8. Some impurities have been added with concentrations in parts per million by weight to show the increase in the neutron source term, over spontaneous fission alone, from the (α, n) reaction in these elements because of the α-decay of plutonium.

The computed gamma ray spectrum is shown in Table *5.9*. The impurities in the plutonium have no effect on these values. A broad maximum exists in the gamma ray spectra at about 75 years, with

Table 5.8. Initial and trace element impurities concentrations

Plutonium density of 19.3 g/cm^3	
Isotope	**Plutonium wt %**
^{238}Pu	0.02
^{239}Pu	93.27
^{240}Pu	6.10
^{241}Pu	0.57
^{242}Pu	0.04
Oxygen	200 ppm
Fluorine	200 ppm
Iron	200 ppm

Table 5.9. Gamma ray spectrum of plutonium at 75 years

Group no.	Energy interval (MeV)	Gamma rays/cm^3/sec
1	8.00 - 10.00	8.11×10^{-1}
2	6.50 - 8.00	3.86×10^{0}
3	5.00 - 6.50	1.99×10^{1}
4	4.00 - 5.00	5.05×10^{1}
5	3.00 - 4.00	1.52×10^{2}
6	2.50 - 3.00	1.72×10^{2}
7	2.00 - 2.50	3.02×10^{2}
8	1.66 - 2.00	5.06×10^{2}
9	1.33 - 1.66	2.68×10^{-1}
10	1.00 - 1.33	1.19×10^{3}
11	0.80 - 1.00	1.97×10^{3}
12	0.60 - 0.80	1.29×10^{5}
13	0.40 - 0.60	6.79×10^{5}
14	0.30 - 0.40	2.26×10^{5}
15	0.20 - 0.30	4.54×10^{5}
16	0.10 - 0.20	6.36×10^{6}
17	0.05 - 0.10	3.55×10^{9}
18	0.01 - 0.05	2.56×10^{9}
	Total	6.12×10^{9}

the values at 50 and 100 years being only slightly less. The time dependence of the gamma ray source spectrum is a result of the build-up and decay of ^{241}Am in the plutonium, with a maximum concentration of 0.51 wt% at about 75 years after the fabrication of pure plutonium with the concentrations in Table 5.8. Examination of the ORIGEN output reveals that at 20 years the ^{241}Am concentration is 0.35 wt% and the total spectrum is at about 75% of the maximum. At 200 years, the ^{241}Am concentration has increased to 0.43 wt% and the total spectrum is 87% of the maximum at 75 years. The initial concentration of ^{238}Pu and ^{241}Pu also affect the time dependence of the gamma ray spectrum, especially at early times.

The specifications for the plutonium from which this sample problem was taken did not indicate the presence of ^{236}Pu, and this isotope was not included in the analysis. It is sometimes erroneously assumed that the ^{236}Pu concentration is so small (on the order of a few parts per billion by weight, ppb) and the half-life so short (2.85 years), that the isotope can be neglected in shielding analyses. At 1 ppb ^{239}Pu, the group six value in Table 5.9 (2.5 - 3.0 MeV) increases by a factor of 20, and at 100 ppb, the value is on the order of 4 × 108 gamma/cm^3/sec. At concentrations on the order of 100 ppb or greater, the thin disk model of configuration number 1 with no self shielding may produce external package dose rates above the regulatory limits. In this case, it will be necessary to use more realistic geometric models and isotopic concentrations. The inclusion of ^{236}Pu in concentrations of several ppb in Table 5.8 will lead to a maximum gamma ray source spectrum in Table 5.9 at an earlier time, about 40 years after production of the plutonium. The maximum value of group six will occur at about 18 years. The inclusion of isotopes such as ^{236}Pu would require that, in addition to source strength calculations, several dose rate calculations be made at several times to determine the maximum package external dose rates. The time of maximum total gamma ray dose is determined by the bottom two groups (0.01 - 0.1 MeV) where the gamma rays are easily attenuated by the package and content materials. The time of maximum exterior dose rates will usually occur between the times of total maximum gamma ray emission and that of maximum value of group six. It is possible that different detectors and conditions (NCT and HAC) could have different times for calculated maximum dose rates. A conservative method requiring only one set of dose rate calculations would be to use a source spectrum with each group value at its maximum, or near maximum, value.

The ORIGEN calculations were run with 27 neutron groups and 18 gamma ray groups. An increase in the same energy range in Table 5.9 may change the spectrum, but the only significant increase will be in the lowest group (below 50 keV). The package external dose rates are unaffected by this energy range. The lower neutron group has no primary source contribution and is included to account for the induced fissions resulting from the spontaneous fissions and (α, n) reactions.

The neutron source spectrum is shown in Table 5.10. The time dependent buildup of the neutron source is the result of the (α, n) source in the trace element fluorine. At the 50-year maximum, the presence of flourine accounts for almost 20% of the total neutron source. The spontaneous fission source drops continuously from a maximum at time zero. The time dependence of the neutron source for the concentrations in Table 5.8 is less than that for the gamma ray source. The 50-year maximum value is less than 5% greater than both the initial spectrum and the 200 year spectrum.

That weapons grade plutonium would contain fluorine in trace amounts as high as 200 ppm is unlikely, and this value has been used in this sample problem only for illustration. However, for any analysis of plutonium shipments or storage, realistic values of trace element concentrations should be included in the neutron source calculations. If the trace element concentrations are reported as maximum or theoretical upper limits, average or nominal values should be used. Assuming that all trace elements appear at maximum concentration simultaneously is unrealistic, leading to overly conservative neutron

Table 5.10. Neutron spectrum of plutonium at 50 years

Energy interval (MeV)		Neutrons/cm^3/sec
6.43 - 20.00		2.33×10^1
3.00 - 6.43		3.82×10^2
1.85 - 3.00		3.72×10^2
1.40 - 1.85		1.78×10^2
0.90 - 1.40		2.38×10^2
0.40 - 0.90		2.56×10^2
0.10 - 0.40		5.27×10^1
0.02 - 0.01		8.02×10^{-1}
	Total	1.50×10^3

dose rate calculations. However, elements with large (α, n) reaction cross sections, such as fluorine and beryllium, should always be included in an analysis if it is possible that they exist in the plutonium.

The ORIGEN-S calculations that produce the time-dependent neutron and gamma sources can also compute the thermal heat source in terms of W/cm^3—the heat generated internally in the plutonium because of its radioactive decay. The time-dependence of the heat generation has the same general feature as the time-dependent gamma ray source, with a maximum of 0.0554 W/cm^3 at about 75 years. Then the maximum heat generation in the plutonium of each inner container in Fig. 5.2 is almost 13 W.

5.8.3 Dose Rate Calculations

The source spectra in Tables 5.9 and 5.10 were input into the MORSE-CG multigroup Monte Carlo for the calculation of the dose rates exterior to the shipping package in Fig. *5.2*. The detector coordinates are given in Table 5.11.

The cross-section library used was the SCALE system 27 neutron-18 gamma-ray multigroup data set discussed in Subsect. 5.6.3. The dose response functions are the 1977 ANSI standard values shown in Tables 5.12 and 5.13. The nuclide densities for the plutonium, steel, and insulation are shown in Table 5.14. Only ^{239}Pu (at 93.9 wt%) and ^{240}Pu (at 6.1 wt%) were used in the transport calculations. All the isotopes in Table 5.8 and those produced in the time-dependent buildup and decay processes have very nearly the same radiation shielding characteristics, with the use of ^{239}Pu being conservative for both gamma ray shielding for the higher-weight isotopes and from induced neutrons in fissile material. The trace elements were not considered. As a further item of conservatism, the plutonium density for the shielding calculations was reduced to 19.0 g/cm^3 from the 19.3 g/cm^3 used in the source generation.

Table 5.11. Detector locations (cm)
(axial reference is shipping drum bottom)

	Top	Bottom	Side	Top	Bottom	Side
NCT		surface			surface	
Height	70.33	-1.0	33.73	169.33	-100.0	33.73
Radius	0.0	0.0	24.29	0.00	0.0	123.29
HAC[a]						
Height				157.18	-89.73	33.73
Radius				0.00	0.00	112.46

[a] Insulation and drum removed.

Table 5.12. ANSI standard neutron flux-to-dose rate conversion factors

Group number	Energy (MeV)	Factor (mrem/hr)/(neutron/sec/cm^2)
1	$2.00 \times 10^{1} - 6.43 \times 10^{0}$	1.4916×10^{-1}
2	$6.43 \times 10^{0} - 3.00 \times 10^{0}$	1.4464×10^{-1}
3	$3.00 \times 10^{0} - 1.85 \times 10^{0}$	1.2701×10^{-1}
4	$1.85 \times 10^{0} - 1.40 \times 10^{0}$	1.2811×10^{-1}
5	$1.40 \times 10^{0} - 9.00 \times 10^{-1}$	1.2977×10^{-1}
6	$9.00 \times 10^{-1} - 4.0 \times 10^{-1}$	1.0281×10^{-1}
7	$4.00 \times 10^{-1} - 1.0 \times 10^{-1}$	5.1183×10^{-2}
8	$1.00 \times 10^{-1} - 1.7 \times 10^{-2}$	1.2319×10^{-2}
9	$1.70 \times 10^{-2} - 3.0 \times 10^{-3}$	3.8365×10^{-3}
10	$3.00 \times 10^{-3} - 3.3 \times 10^{-4}$	3.7247×10^{-3}
11	$5.50 \times 10^{-4} - 1.0 \times 10^{-4}$	4.0150×10^{-3}
12	$1.00 \times 10^{-4} - 3.0 \times 10^{-5}$	4.2926×10^{-3}
13	$3.00 \times 10^{-5} - 1.0 \times 10^{-5}$	4.4744×10^{-3}
14	$1.00 \times 10^{-5} - 3.0 \times 10^{-6}$	4.5676×10^{-3}
15	$3.05 \times 10^{-6} - 1.7 \times 10^{-6}$	4.5676×10^{-3}
16	$1.77 \times 10^{-6} - 1.3 \times 10^{-6}$	4.5185×10^{-3}
17	$1.30 \times 10^{-6} - 1.1 \times 10^{-6}$	4.4879×10^{-3}
18	$1.13 \times 10^{-6} - 1.0 \times 10^{-6}$	4.4665×10^{-3}
19	$1.00 \times 10^{-6} - 8.0 \times 10^{-7}$	4.4345×10^{-3}
20	$8.00 \times 10^{-7} - 4.0 \times 10^{-7}$	4.3271×10^{-3}
21	$4.00 \times 10^{-7} - 3.2 \times 10^{-7}$	4.1975×10^{-3}
22	$3.25 \times 10^{-7} - 2.2 \times 10^{-7}$	4.0976×10^{-3}
23	$2.25 \times 10^{-7} - 1.0 \times 10^{-7}$	3.8390×10^{-3}
24	$1.00 \times 10^{-7} - 5.0 \times 10^{-8}$	3.6748×10^{-3}
25	$5.00 \times 10^{-8} - 3.0 \times 10^{-8}$	3.6748×10^{-3}
26	$3.00 \times 10^{-8} - 1.0 \times 10^{-8}$	3.6748×10^{-3}
27	$1.00 \times 10^{-8} - 1.0 \times 10^{-11}$	3.6748×10^{-3}

Table 5.13. ANSI standard photon flux-to-dose rate conversion factors

Group number	Energy (MeV)	Factor (mrem/hr)/(photon/sec/cm^2)
1	10.00 — 8.00	8.771×10^{-3}
2	8.00 — 6.50	7.478×10^{-3}
3	6.50 — 5.00	6.374×10^{-3}
4	5.00 — 4.00	5.413×10^{-3}
5	4.00 — 3.00	4.622×10^{-3}
6	3.00 — 2.50	3.959×10^{-3}
7	2.50 — 2.00	3.468×10^{-3}
8	2.00 — 1.66	3.019×10^{-3}
9	1.66 — 1.33	2.627×10^{-3}
10	1.33 — 1.00	2.205×10^{-3}
11	1.00 — 0.80	1.832×10^{-3}
12	0.80 — 0.60	1.522×10^{-3}
13	0.60 — 0.40	1.172×10^{-3}
14	0.30 — 0.30	8.759×10^{-3}
15	0.30 — 0.20	6.306×10^{-3}
16	0.20 — 0.10	3.833×10^{-3}
17	0.10 — 0.05	2.669×10^{-3}
18	0.05 — 0.01	9.347×10^{-3}

Table 5.14. Nuclide densities

Material	Density (g/cm)	Nuclides	Weight (%)	Density (atoms/barn-cm)
Plutonium	19.000	^{239}Pu	93.9	4.496(-2)
		^{240}Pu	6.1	2.909(-3)
Insulation	0.255	H	6.2	9.447(-3)
		C	44.4	5.683(-3)
		O	49.4	4.742(-3)
Stainless steel	7.920	Cr	19.0	1.743(-3)
		Mn	2.0	1.736(-3)
		Fe	69.5	5.936(-2)
		Ni	9.5	7.721(-3)

The calculated dose rates in mrem/h for the detector locations in Fig. 5.2 and Table 5.11 are listed in Table 5.15. These dose rates are for configuration number one, the thin disk plutonium cylinders that minimize the self-shielding effect. The CFR limits for transportation packages are also given in Table 5.15. The neutron dose rates also include the contribution from secondary gamma rays generated from neutron interactions, which is less than one percent of the neutron dose rate in all cases. The surface dose rates for configuration number 2 are shown in Table 5.16. Here the plutonium is lumped as a square cylinder in each inner container, which has the effect of greatly self-shielding the gamma rays but also greatly increases the induced fission neutrons in the fissile material.

The third geometric configuration investigated was for one inner container alone removed from the shipping package with the thin disk plutonium shape of configuration number one. This configuration would be for long-term storage of the plutonium after the inner containers were moved from the shipping packages. The maximum surface dose rate on the container bottom was calculated as 382.3 ± 8.3% mrem/h. Approximately one-fourth of this dose rate was the result of neutrons.

5.8.4 ALARA Concepts

The dose rates calculated for the three configurations are near the maximum possible for the mass, shape, and orientation of the inner containers; the maximum gamma ray dose rate for configuration numbers one and three; and the maximum neutron dose rate for configuration number two. If the thin disk shapes were vertically orientated instead of horizontal, the maximum dose rates for configurations numbers one and three would be on the side instead of the bottom, with the side and bottom values in Table 5.15 approximately reversed. If the plutonium were moved from a central location in the inner container closer to an interior surface, or if the inner container were closer to the surface of the shipping

Table 5.15. Calculated maximum dose rates for configuration one (mrem/h). The uncertainties are one standard deviation as a percent of the mean values.

	Package surface			One meter from surface of package [a, b]		
	Side	Top	Bottom	Side	Top	Bottom
Normal Conditions of Transport						
Photon	64.7 ± 2.5%	29.0 ± 5.8%	85.8 ± 6.8%	3.12 ± 2.6%	1.51 ± 4.2%	2.58 ± 2.2%
Neutron	10.6 ± 0.8%	7.7 ± 2.3	17.1 ± 2.5	0.43 ± 0.6%	0.33 ± 0.7%	0.44 ± 0.6%
Total	75.3 ± 2.2%	36.6 ± 4.6%	102.8 ± 5.7%	3.55 ± 2.3%	1.84 ± 3.4%	3.02 ± 1.9%
10 CFR 71 limit	200	200	200	10	10	10
Hypothetical Accident Conditions						
Photon	NA [c]	NA	NA	8.02 ± 1.9%	2.36 ± 0.9%	4.88 ± 1.3%
Neutron	NA	NA	NA	0.51 ± 0.6%	0.53 ± 0.4%	0.62 ± 0.5%
Total	NA	NA	NA	8.53 ± 1.8%	2.89 ± 0.7%	5.50 ± 1.2%
10 CFR 71 limit	NA	NA	NA	1000	1000	1000

[a] Drum exterior for Normal Conditions of Transport.
[b] Inner container exterior for Hypothetical Accident Conditions.
[c] NA—Not Applicable.

drum, some of the exterior dose rates would increase. However, the packing material is assumed to keep the centrally located placement of the plutonium and inner container intact, at least during NCT.

From Tables 5.15 and 5.16 one can see that the shipping package dose rates investigated in these sample problems are well within the regulatory limits. But from an ALARA aspect, it may be necessary to reduce these dose rates, especially for the long-term storage of the inner containers. Several of these containers stored together may violate 20 CFR or local installation dose rate limits.

Shielding could be added around the plutonium inside the inner container, around the inner container inside the shipping drum, or exterior to the entire package. Common gamma ray shielding materials are steel, lead, tungsten, and depleted uranium, in increasing order of material density (weight) and shielding effectiveness. Other than steel, each material presents some difficulties: lead "slump", fabrication problems with tungsten and depleted uranium, and the small amount of radioactivity from uranium. Adding any additional material inside the inner container would be the most efficient method of shielding, but criticality safety considerations may preclude this. Additional material outside any of the containers could greatly increase the shipping package weight, although this would seem acceptable for long-term storage. An alternative would be to ship and store less plutonium per container. The shipment and storage of radioactive material under the control of the DOE weapons programs will increase in volume and complexity for the foreseeable future, and the ALARA concepts applied to these programs will require more study than the current methods of ensuring that the shipping package dose rates are below the 10 CFR 71 limits.

Table 5.16. Calculated maximum dose rates (mrem/h)

	Package surface		
	Side	Top	Bottom
Normal Conditions of Transport			
Photon	5.5 ± 6.2%	7.00 ± 8.4%	8.74 ± 6.2%
Neutron	24.8 ± 2.0%	21.3 ± 4.4%	30.3 ± 2.7
Total	30.2 ± 2.5%	28.3 ± 5.0%	39.0 ± 3.1%
10 CFR 71 limit	200	200	200

5.8.5 Calculational Modeling

The surface detector locations were modeled in this example as being 1 cm off the surface because of the possible high statistical uncertainty inherent in Monte Carlo point detectors located in or on the surface of a scattering material. However, the statistical uncertainty reported for these 1-cm detectors should encompass the actual surface dose rate. It is possible to calculate an actual surface dose rate averaged over some area, but as this area is made smaller to find the maximum dose rate, the statistical uncertainty may become unacceptably large. It is possible to calculate a surface dose rate with a discrete ordinates code over a very small area (there is no statistical uncertainty here), and many shipping package designs are amenable to solution by this method. However, the geometric modeling of some complex package contents may require much simplification with discrete ordinates codes. The point-kernel method can accurately model a point surface detector for gamma rays, but this method is not applicable for neutron calculations. The locations of maximum exterior dose rates may not be apparent for some package designs, and a few preliminary calculations may be necessary to approximate these locations. For the geometry in Figure 5.2, the maximum dose rates on the top and bottom of the package are assumed to be on the axial centerline of the package, and the maximum side dose rate is at an axial location corresponding to the mid-point between the two plutonium pieces. If there were only one inner container with the thin disk plutonium, the package side locations of maximum dose rate would be somewhere above and below the axial location of the plutonium disk because of the self-shielding of the plutonium in the radial direction.

A final modeling comment involves the Monte Carlo gamma ray dose rate calculation from a spectrum such as that in Table 5.9. Except for configuration number three (the inner container alone), the bottom two spectral values (100 keV and below) contribute very little to the calculated dose rates, although they make up most of the total source strength. The higher-energy spectra must be adequately sampled to calculate realistic dose rates, but this sampling cannot be done if a single, unbiased sampling is made for the entire spectra. Various biasing schemes are available, but a simpler method is to divide the energy range into intervals where the spectral values are the same order of magnitude and make a separate calculation for each interval with its own source strength, combining all of the separate results for the final dose rates. The extreme case for this method would be to make a separate calculation for each spectral energy interval.

The shipment and storage of packages with only uranium as the radioactive material will generally give rise to dose rate levels much lower than that for plutonium, sometimes not much greater than background levels. The principle item of concern is the assumed concentration, in parts per billion (ppb), of the ^{232}U isotope in enriched uranium. At a maximum assumption of 40 ppb ^{232}U by weight in enriched uranium (much more than would ever occur in actual uranium — less than 5 ppb is more realistic), calculated dose rates exterior to packages with other conservative assumptions could exceed 100 mrem/h. However, it is unlikely that a uranium contents shipping package measurement would ever exceed 10 mrem/h. Whatever the dose rate, calculated or measured, it will be almost entirely from gamma rays. Little neutron radiation from uranium would be detectable on a package exterior. As a simple and extremely conservative neutron dose rate calculation, one can assume that all of the uranium is located at a point (no self-shielding) in the package that has the closest distance r to the package exterior surface for any uranium. All package materials are ignored, and a void flux calculation, $1/4\pi r^2$, is made for each source energy group. This flux is multiplied both by the response function for that group and by the neutron multiplication factor $(1\text{-}k_{eff})^{-1}$. The sum over groups will give a dose rate that in many cases will not be much above background and should never exceed a few mrem/h.

5.9 QUALITY ASSURANCE

Quality assurance (QA) activities for all related packaging activities, including radiation shielding aspects, must conform with the applicable requirements of DOE Order 5700.6C, 10 CFR 71, Subpart H, and other relevant codes or standards.

The selective application of QA requirements begins with the adherence to engineering procedures for the control of all activities during the design of the packaging. These approved procedures typically include control of design input, data, and assumptions; control of documents, records, change, software, and interface controls; and design verification.

A nonconformance and corrective action system should be in place to handle deviations or nonconformances identified during the design phase. Deviations from requirements and procedural controls should be documented and appropriate personnel identified to evaluate and disposition each deviation adequately. A record-keeping system should be established because records of the design must be maintained according to approved procedures.

Periodic internal assessments of the adequacy of the design control systems should be accomplished by the Engineering organization to ensure the effectiveness of these controls.

THIS PAGE INTENTIONALLY LEFT BLANK

5.10 REFERENCES

1. U.S. Department of Energy, *Packaging and Offsite Transportation of Components, Special Assemblies, and Radioactive Material Associated with the Nuclear Weapons Program,* DOE Order 5610.12, Office of the Deputy Assistant Secretary for Military Applications, Washington, D. C. (Draft 5), June 1990.

2. U.S. Nuclear Regulatory Commission, *Packaging and Transportation of Radioactive Materials*, Code of Federal Regulations, Title 10, Part 71, (Federal Register, Vol. *53,* No. 110, p. 21556), Office of the Federal Register, National Archives and Records Administration, Washington, D.C., June 8, 1988.

3. International Atomic Energy Agency, IAEA Safety Series No. 6, *Regulations for the Safe Transport of Radioactive Material,* (1985 Edition), Vienna, 1988.

4. U.S. Nuclear Regulatory Commission, *Standard Format and Content of Part 71 Applications for Approval of Packaging for Radioactive Material,* Regulatory Guide 7.9, Rev. 2, Washington, D.C., 1986.

5. U.S. Department of Energy, *Packaging Review Guide for Reviewing Safety Analysis Report for Packages,* 1987.

6. American National Standards Institute, *American National Standard Radiation Protection Instrumentation Test and Calibration,* ANSI N323-1978, 1978.

7. Parks, C. V., et al., *Assessment of Shielding Analysis Methods, Codes and Data for Spent Fuel Transport/Storage Applications*, ORNL/CSD/TM-246, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory, July 1988. (Also, see *The Packaging Handbook*, L. B. Shappert, ed., Chapter 9, *Shielding*, ORNL, in preparation.)

8. Bell, M. J., *ORIGEN - The ORNL Isotope Generation and Depletion Code,* ORNL-4628, Union Carbide Corp., Nuclear Division, Oak Ridge National Laboratory, 1973.

9. Kinji Koyama, Naoki Yamano, and Shun-ich Miyasaka, *ORIGEN-JR: A Computer Code for Calculating Radiation Sources and Analyzing Nuclide Transmutations,* JAERI-M-8229, Japan Atomic Energy Research Institute, May 1979.

10. Fischer, U. and Wiese, H. W., *improved and Consistent Determination of the Nuclear Inventory of Spent PWR Fuel on the Basis of Cell-Burnup Methods Using KORIGEN,* ORNL-tr-5043 (KfK 3814), January 1983.

11. *ORJGEN2. 1: Isotope Generation and Depletion Code Matrix Exponential Method,* CCC-371, Available from Radiation Shielding Information Center at Oak Ridge National Laboratory.

12. Hermann, O. H. and Westfall, R. M., *ORIGEN-S: Scale System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms,* as described in Sect. F7 of *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, Vols. 1-3,* NUREG/CR-0200, Rev. 4 (ORNL/NUREG/CSD-2/R4), Vols. I, II, and III (draft February 1990). Available from Radiation Shielding Information Center, ORNL, as CCC-545.

13. Ford, W. E., III, et al., *FCXSEC: Multigroup Cross-Section Libraries for Nuclear Fuel Cycle Shielding Calculations,* ORNL/TM-7038, Union Carbide Corp., Nuclear Division, Oak Ridge National Laboratory, May 1980.

14. England, T. R., *Cinder - A One-Point Depletion and Fission Product Program,* WAPD-TM-334, Westinghouse Electric Corporation, 1962, Rev. 1964.

15. England, T. R., Wilson, W. B., and Stamatelatos, M. G., *Fission Product Data for Thermal Reactors, Part 2: Users Manual for EPRI-CINDER Code and Data,* EPRI NP-356 Part 2, LA-6746-MS, Electric Power Research Institute, December 1976.

16. Wilson, W. B., et al., *Development of the SOURCES Code and Data Library for the Calculation of Neutron Sources and Spectra Form (α, n) Reactions, Spontaneous Fission, and β Delayed Neutrons,* LA-9841-PR, p. 65, Los Alamos National Laboratory, August 1983.

17. Gumprecht, R. O., *Mathematical Basis of Computer Code RIBD,* DUN-4136, Douglas-United Nuclear, Inc., June 1968.

18. Engle, R. L., Greenborg, J., and Hendrickson, M. M., *ISOSHLD - A Computer Code for General Purpose Isotope Shielding Analysis,* BNWL-236, Brookhaven National Laboratory (June 1966) and Supplement (March 1967).

19. *BREMRAD: External and Internal Bremsstrahlung Calculation Code,* CCC-31, available from Radiation Shielding Information Center at Oak Ridge National Laboratory.

20. Hermann, O. W., *SAS2: A Coupled One-Dimensional Depletion and Shielding Analysis Module*, as described in Sect. S2 of *SCALE: A Modular Code System for Performing Standardized Computer analyses for Licensing Evaluation, Vols. 1-3,* NUREG/CR-0200, Rev. 4, (ORNL-NUREG/CSD-2/R4), Vols. I, II, and III (draft February 1990). Available from Radiation Shielding Information Center, ORNL, as CCC-493/GADCGGP.

21. *QAD-CGGP: A Combinatorial Geometry Version of QAD-P5A, A Point Kernel Code System for Neutron and Gamma-Ray Shielding Calculations Using the GP Buildup Factor,* Available from Radiation Shielding Information Center, ORNL, as CCC-493/GADCGGP.

22. Trubey, D. K., *New Gamma-Ray Buildup Factor Data for Point Kernel Calculations: ANS-6.4.3, Standard Reference Data,* NUREG/CR-5740 (ORNL-RSIC-49/R1), Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory, August 1991.

23. Grove Engineering, Inc., *Microshield User's Manual, A Program for Analyzing Gamma Radiation Shielding, Version 2.0,* Washington Grove, Maryland, 1986.

24. Su, S., Baylor, K. J., and Engholm, B. A., *PATH Gamma Shielding Code User's Manual,* GA-A16772, GA Technologies, September 1982.

25. Engle, Ward W., Jr., *A User's Manual for ANISN, A One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering,* K-1693, Union Carbide Corp., Nuclear Division, Oak Ridge Gaseous Diffusion Plant, updated June 1973.

26. Greene, N. M. and Craven, C. W., Jr., *XSDRN: A Discrete Ordinates Spectral Averaging Code,* ORNL/TM-2500, Union Carbide Corp., Nuclear Division, Oak Ridge National Laboratory, July 1969.

27. Greene, N. M. and Petrie, L. M., *XSDRNPM-S: A One-Dimensional Discrete-Ordinates Code for Transport Analysis,* Vol. 2, Sect. F3 of *SCALE: A Modular Code System for performing Standardized ComputerAnalyses forLicensing Evaluation, Vols. 1-3,* NUREG/CR-0200, Rev. 4, ORNL/NUREG/CSD-2/R4, Vols. I, II, and III (draft February 1990). Available from Radiation Shielding Information Center, ORNL, as CCC-545.

28. *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, Vols. 1-3,* NUREG/CR-0200, Rev. 4, ORNL/NUREG/CSD-2/R4, Vols. I, II, and III(draft February 1990). Available from Radiation Shielding Information Center, ORNL, as CCC-545.

29. O'Dell, R. Douglas, Brinkley, Forest W., Jr., and Marr, Duane R., *User's Manual for ONEDANT: A Code Package for One-Dimensional Diffusion-Accelerated, Neutral-Particle Transport,* LA-6941-MS, Los Alamos National Laboratory, February 1982.

30. O'Dell, R. Douglas and Alcouffe, Raymond E., *Transport Calculations for Nuclear Analyses: Theory and Guidelines for Effective Use of Transport Codes,* LA-10983-MS, Los Alamos National Laboratory, September 1987.

31. Rhoades, W. A. and Childs, R. L., *The DORT Two-Dimensional Discrete Ordinates Code,* Nucl. Sci. Eng., 99, 88-89 (May 1988).

32. Rhoades, W. A. and Childs, R. L., *The TORT Three-Dimensional Discrete-Ordinates Neutron/Photon Transport Code,* ORNL-6268, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory, November 1987.

33. Lathrop, K. D., *THREETRAN:A Program to Solve the Multigroup Discrete Ordinates Transport Equation in (x,y,z) Geometry,* LA-633-MS, Los Alamos National Laboratory, May 1976.

34. Baker, V. and Childs, R. L., *FALSTF: Calculator of Response of Detectors External to Shielding Configuration,* Available as package CCC-351 of the Radiation Shielding Information Center at Oak Ridge National Laboratory, 1979.

35. *SPACETRAN III Dose Calculations at Detectors at Various Distances From the Surface of a Cylinder, I: Integration of ANISN Leakage Data, II: Assumed Powers of Cosine Angular Distribution, III: Integration of DOT III Scalar Flux Data,* available from Radiation Shielding Information Center at Oak Ridge National Laboratory as CCC-120.

36. Bucholz, J. A., *XSDOSE: A Module for Calculating Fluxes and Dose Rates at Points Outside a Shield,* Vol., Sect. F4 of *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation,* Vols. 1-3, NUREG/CR-0200, Rev. 4, ORNL/NUREG/CSD-2/R4, Vols. I, II, and III (draft February 1990). Available from Radiation Shielding Information Center, ORNL, as CCC-545.

37. Cramer, S. N., *Applications Guide to the MORSE Monte Carlo Code,* ORNL/TM-9355, 1985.

38. Emmett, M. B., *MORSE-CGA, A Monte Carlo Radiation Transport Code with Array Geometry Capabiliiy*, ORNL-6174, Martin Marietta Energy Systems, Inc, Oak Ridge National Laboratory, April 1985.

39. West, J. T., Hoffman, T. J., and Emmett, M. B., *MORSE-SGC/S for the SCALE System,* Vol.2, Sect. F9 of *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation,* Vols. 1-3, NUREG/CR-0200, Rev. 4, ORNL/NUREG/CSD-2/R4, Vols. I, II, and III (draft February 1990). Available from Radiation Shielding Information Center, ORNL, as CCC-545.

40. West, J. T. and Emmett, M. B., *MARS: A Multiple Array System Using Combinatorial Geometry,* Vol - 3, Sect. M9 of *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation,* Vols. 1-3, NUREG/CR-0200, Rev. 4, ORNL/NUREG/CSD-2/R4, Vols. I, II, and III (draft February 1990). Available from Radiation Shielding Information Center, ORNL, as CCC-545.

41. Los Alamos Monte Carlo Group, *MCNP—A General Monte Carlo Code for Neutron and Photon Transport Version 3A*, LA-7396-M, Rev. 2, Los Alamos National Laboratory, September 1986.

42. Lichtenstein, H., et al., *The SAM-CE Monte Carlo System for Radiation Transport and Criticality Calculations in Complex Configurations,* (Rev. 7.0), MAGI report MR-705207, also EPRI CCM-8 (1980).

43. Nimal, J. C., et al., *Programme de Monte Carlo Polycinétiqueà) Trois Dimensions TRIPOLI,* CEA.N.1919, Saclay (1976); also CCC-272, Oak Ridge National Laboratory.

44. Plechaty, E. F. and Kimlinger, J. R., *TART NP: A Coupled Neutron-Photon Monte Carlo Transport Code,* UCRL-50400, Vol. 14, Lawrence Livermore National Laboratory (1976).

45. Wilcot, T. P. and Lent, E. M., *COG - A Particle Transport Code Designed to Solve the Boltzmann Equation for Deep-Penetration (Shielding) Problems,* M-221-1, Lawrence Livermore National Laboratory (1989).

46. Straker, E. A., Morrison, G. W., and Odegaarden, R. H., *A Coupled Neutron and Gamma-Ray Cross Section Library for Use in Shielding Calculations,* Trans. Am. Nucl. Soc., 15, 535, (1972).

47. Bartine, D. E., et al., *Production and Testing of the DNA Few Group Cross Section Library*, ORNL/TM-4840, Union Carbide Corp., Nuclear Division, Oak Ridge National Laboratory, 1975.

48. Barrett, R. J. and MacFarlane, R. E., *Coupled Neutron and Photon Cross Sections for Transport Calculations,* LA-7808-MS, Los Alamos National Laboratory, 1979.

49. *BUGLE-80 Coupled 47 Neutron, 20 Gamma-ray,* P_3 *Cross Section Library for LWR Shielding Calculations by the ANS-6.1.2 Working Group on Multi group Cross Sections,* available from Radiation Shielding Information Center, ORNL, as DLC-75.

50. Simmons, G. L., et al., *Analysis of the Browns Ferry Unit 3 Irradiation Experiments, NP-3719,* Research Project 827-1, Science Applications, Inc., November 1984.

51. Broadhead, B. L., Parks, C. V., and Pope, R. B., *Assessment of Proposed Dose Factor Changes to Shipping Cask Design and Operations,* 1992 International High-Level Radioactive Waste Management Conference, Las Vegas, Nevada, April 12-15, 1992.

THIS PAGE INTENTIONALLY LEFT BLANK